1051-A-11 Y3.N88:25/4485/v.2
SEP 02 1987

NUREG/CR-4485
PNL-5606
Vol. 2

The Impact of Fuel Cladding Failure Events on Occupational Radiation Exposures at Nuclear Power Plants

Case Study: PWR During an Outage

Prepared by M. P. Moeller, G. F. Martin, J. L. Kenoyer

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

COMPLETED

ORIGINAL

BEST COPY AVAILABLE

NOTICE

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability of responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights.

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082,
   Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Information Support Services, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

BEST COPY AVAILABLE

I

NUREG/CR-4485
PNL-5606
Vol. 2
R1

The Impact of Fuel Cladding Failure Events on Occupational Radiation Exposures at Nuclear Power Plants

Case Study: PWR During an Outage

Manuscript Completed: December 1986
Date Published: August 1987

Prepared by
M. P. Moeller, G. F. Martin, J. L. Kenoyer

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Reactor Accident Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN B2987

BEST COPY AVAILABLE

II

ABSTRACT

This report is the second in a series of case studies designed to evaluate the magnitude of increase in occupational radiation exposures at commercial U.S. nuclear power plants resulting from small incidents or abnormal events. The event evaluated is fuel cladding failure, which can result in elevated primary coolant activity and increased radiation exposure rates within a plant. For this case study, radiation measurements were made at a pressurized-water reactor (PWR) during a maintenance and refueling outage. The PWR had been operating for 22 months with fuel cladding failure characterized as 105 pin-hole leakers, the equivalent of 0.21 percent failed fuel. Gamma spectroscopy measurements, radiation exposure rate determinations, thermoluminescent dosimeter (TLD) assessments, and air sample analyses were made in the plant's radwaste, pipe penetration, and containment buildings.

Based on the data collected, evaluations indicate that the relative contributions of activation products and fission products to the total exposure rates were constant over the duration of the outage. This constancy is due to the significant contribution from the longer-lived isotopes of cesium (a fission product) and cobalt (an activation product). For this reason, fuel cladding failure events remain as significant to occupational radiation exposure during an outage as during routine operations. As documented in the previous case study (NUREG/CR-4485 Vol. 1), fuel cladding failure events increased radiation exposure rates an estimated 540 percent at some locations of the plant during routine operations. Consequently, such events can result in significantly greater radiation exposure rates in many areas of the plant during the maintenance and refueling outages than would have been present under normal fuel conditions.

BLANK PAGE

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has the responsibility to monitor and regulate the operation of commercial U.S. nuclear power plants. When operating practices, safety issues, or regulatory goals indicate that a regulatory requirement may need to be changed, NRC considers both onsite and offsite radiation protection before making a decision. At present, there are a significant number of existing and potential generic issues involving regulatory decisions that focus on small incidents or abnormal events. Unlike events of a greater magnitude, such as severe core damage, small incidents are not generally expected to result in any increase in radiation exposure to the general public. Therefore, from a radiation protection standpoint, the radiological consequences of such events are limited primarily to their impact on the workers within the nuclear power plant. In order to resolve the numerous generic issues pertaining to small incidents, it is first necessary to quantify the significance of the potential onsite radiological consequences.

This report is the second in a series of case studies designed to evaluate the magnitude of increases in occupational radiation exposures at commercial U.S. nuclear power plants resulting from small abnormal events. The incident evaluated in each case study is fuel cladding failure. Although this event may have no measurable impact on the general public, it can result in elevated primary coolant activity and increased radiation exposure rates at some locations within a plant. This evaluation of the radiological impact of the fuel cladding failure events on nuclear power plant workers is intended to serve as an indicator of the significance of small incidents or abnormal events in general. The resulting information should be useful in the probabilistic risk assessment of the consequences of such events, and such assessments should, in turn, provide data to support the regulatory decision process.

The first case study documented the impact of fuel cladding failure at a pressurized-water reactor (PWR) during routine operations (NUREG/CR-4485 Vol. 1, PNL-5606 Vol. 1). The results of that study indicated that fuel cladding failure events increased radiation exposure rates an estimated 540 percent at some locations of the plant during routine operations. Consequently, it was suggested that such events could result in significantly greater radiation exposure rates during the subsequent maintenance and fuel handling activities than would have been present under normal fuel conditions.

The purpose of this report is to evaluate the impact of failed fuel at a PWR on occupational exposure rates during the plant's subsequent maintenance and refueling outage. To accomplish the task objectives, measurements were made during the next maintenance and refueling outage at the PWR referenced in NUREG/CR-4485 Vol. 1 that had operated with fuel having excessive cladding failure. Measurements that had been taken in areas of routine worker access while the plant was operating were repeated under the outage conditions and new measurements were taken inside the plant's containment building and spent fuel pool pump room.

An unexpected event that occurred at the PWR during the maintenance and refueling outage had a significant impact on the assessment of the consequences of fuel cladding failure events. As part of the refueling operation, fuel

assemblies with leaking rods were to be replaced. Prior to being tested for leaking rods, the fuel assemblies were transferred to the spent fuel pool. During the testing process, a portion of a fuel rod identified as leaking was noted to be broken and a piece of it appeared to crumble. Small, highly-radioactive fuel particles were released into the pool. Subsequent maintenance and refueling activities resulted in some of these particles being picked up on tools and transported throughout the fuel handling building. Fortunately, from a radiation protection standpoint, the particles did not appear to become airborne for significant periods of time. Fuel particles were not observed in any air monitoring surveys or in any ventilation system filters during this study. The dispersal appeared to have been limited to the Fuel Handling Building floor and locations of apparent worker contact such as hand-railings and door knobs. The cost associated with containing and removing these particles, and the occupational exposures resulting from their presence significantly impact the consequences of fuel cladding failure events.

In order to apply the results of this task to risk assessments of generic issues, it is necessary to consider the data within the context of the plant and fuel characteristics of the PWR having fuel with excessive cladding failure. When it entered the maintenance and refueling outage, the unit had been operating for approximately twenty-two months since its initial fuel load. Measurements were taken on the eighty-first and the eighty-second days of the outage. The PWR is power rated as a 3560 megawatt-thermal (1100 megawatt-electric [MWe]) unit. When NUREG/CR-4485 Vol. 1 was completed in November 1985, the best available estimate of the fuel cladding failure events characterized the degraded fuel as 62 pin-hole leakers, which translates to 0.13 percent failed fuel. This estimate suggests that the level of degraded or failed fuel was more than ten times that of the normal fuel operating conditions, although that level is still within the plant's technical specifications. As a result of detailed inspection and testing of individual assemblies of the core in December 1985 and January 1986, the final estimate on the number of leaking fuel rods was 105. These rods were contained in 12 fuel assemblies. Considering that there are a total of 49580 fuel rods within the reactor core, the 105 pin-hole leakers translate to 0.21 percent failed fuel.

There is another plant characteristic affecting the application of these data to risk assessment. As a result of elevated primary coolant activity, the pressurizer steam space sample line in the PWR with failed fuel was modified to provide for continuous venting of the steam space. This change was made to enhance the removal of fission product gases from the reactor coolant system. The impact of this modification during operation was to reduce the equilibrium coolant activity levels by an estimated 30 to 40 percent. When considering how the data from this study could be applied to a PWR that does not have the pressurizer steam space vent (off-gas) line, it is necessary to determine which plant locations, systems and components would experience greater radiation exposure rates at the beginning of the outage as a result of higher coolant activity and radiation levels from the steam space effluent.

The identification of appropriate measurement locations was based on several factors, including measurement locations used for the PWR "During Routine Operations" task, areas of worker activity during the maintenance and

refueling outage, areas near major system components within containment, areas throughout the primary coolant system loop, and areas within the spent fuel pool pump room. Using the locations where measurements were taken while the unit was operating facilitated both the comparison of the routine operations gamma spectra with the outage gamma spectra and the analysis of the radionuclide compositions. In addition, this approach provided an indication of the changing source term with the decay of each radionuclide. Major system components of interest within containment included the pressurizer and steam generators.

Measurements were performed using four basic techniques. First, an intrinsic germanium (Ge) detector was used to distinguish the energy of the gamma photons through spectral analyses. Second, at the same time and location of the gamma measurements, exposure rates were determined using a portable ionization chamber. Third, at some gamma measurement locations where significant beta contributions were expected, eight-element thermoluminescent dosimeters (TLDs) were exposed for a timed interval. Finally, near measurement locations where airborne radionuclide concentrations were suspected, air samples were collected with a high-efficiency filter and silver-zeolite cartridge for analyses of particulates and iodine, respectively.

As a result of these measurements, it was possible to determine the percentage of the exposure rate at each measurement location attributable to each radionuclide identified through the gamma spectrum analyses. The categories of radionuclides included fission products, activation products, and other means of gamma photon and x-ray production including those radionuclides occurring naturally. The ability to distinguish the radiation exposure contribution of fuel cladding failure from that of corrosion product buildup and other sources was the key to determining whether fuel cladding failure events significantly impact the exposure rates in the plant. The TLD measurements were designed to quantify the beta contribution to the exposure rates, and the air samples were used to estimate the potential contribution to occupational exposure via inhalation.

In order to determine the primary contributors to exposure rates, measurements were taken at strategic locations in the piping systems of both the chemical and volume control system and the primary reactor coolant system. Although these systems were not in service during the outage, surveys confirmed that the primary radiation sources in the plant buildings were radionuclides trapped within the components and piping comprising these systems. Examination of radiation survey maps and discussions with plant health physicists and engineers provided information to correlate specific areas of outage activity with significant exposure rates. With the aid of plant personnel, the primary radiation source in each area selected for measurement was identified.

To facilitate a limited comparison, selected measurements taken at the PWR during routine operations were repeated during the outage. These included five measurements in the auxiliary building and one in the pipe penetration building. In each case, care was taken to duplicate the original measurement setup as precisely as possible. During routine operations, high radiation exposure rates and elevated concentrations of airborne contamination had prohibited taking measurements inside the containment building. Consequently,

there were no previous containment measurements for comparison. During routine operations, radiation levels in the spent fuel pool and associated spent fuel pool pumps, heat-exchangers, and recirculation piping had not been affected by the fuel cladding failure. However, it was determined that exposure rates in these areas had increased as a result of the refueling operation. Data on these increases were thus significant in quantifying the impact of the cladding failure on occupational exposures. As was the case with the containment building measurements, there was no previous information to compare with the measurements taken near the spent fuel pool systems and recirculation piping.

Analysis of the data is intended to provide information on radiation exposure rates in specific areas of the plant as well as three functional categories of plant systems and components. Measurements were made in nineteen separate locations and categorized according to the chemical and volume control system, the primary coolant system and components, and the refueling activities and spent fuel storage. In addition, individual radionuclide contributions to the exposure rates are evaluated in two ways. First, the radionuclide contributions to exposure rates are evaluated according to the individual radionuclides. The unique decay rate of each radionuclide results in a time-dependent contribution to exposure rate following reactor shutdown. Second, the radionuclide contributions to exposure rates are evaluated according to the contribution of activation versus fission products to determine their relative importance as a funtion of time.

Evaluation of the gamma photon spectral measurements included identification of radionuclides and analyses of the photon peaks to determine their individual contributions to exposure rates. Analyses of the gamma spectra began with the identification of all significant photon peaks using a Gaussian fit model. Many other less significant peaks were identified by a manual method. The radionuclide contributing to each peak was then identified by its characteristic energy and half-life. This procedure constituted the qualitative analyses. Analyses of the gamma spectral measurement data entered the quantitative stage with the application of a computer program to calculate the radiation exposure rate attributable to each radionuclide peak.

Before presenting the gamma photon spectral measurement data and associated radiation exposure estimates, it is noteworthy to review the TLD exposures and air sampling data. The TLD data indicate that there was no significant contribution from beta radiation to occupational radiation exposures at the measured locations. This was evident because the readouts on the chip with a thin mylar covering and the chip with a thicker 125-mil (0.125 in.) aluminum covering are equivalent. In addition to determining the lack of a beta radiation contribution to exposure rate, the TLD exposures were used as an indicator of the gamma photon field in support of the ionization chamber measurements. Analyses of the filter papers and silver-zeolite cartridges from collected air samples indicated that airborne radionuclides were not present at any of the measurement locations. This result is consistent with the routine air sample data collected by the plant licensee during the outage.

The results of the gamma photon spectral measurements have been analyzed according to the relative radionuclide contributions to plant systems. This

analysis indicates that there exists:

- a greater contribution to exposure rate from activation products in the refueling activities and spent fuel storage than in the chemical and volume control system;
- a greater contribution to exposure rate from activation products in the primary coolant system and compontents than in the chemical and volume control system;
- a greater contribution to exposure rate from Co-58 in areas associated with the spent fuel storage than elsewhere in the plant;
- a greater contribution to exposure rate from the 511 keV annihilation peak in the refueling activities and spent fuel storage than in the primary coolant system and components or in the chemical and volume control system (This appears to be related to the significant amount of Co-58, which is a positron emitter, in areas associated with the spent fuel storage);
- a greater contribution to exposure rate from Nb-95 and Zr-95 in the refueling activities and spent fuel storage than in the primary coolant system and components or in the chemical and volume control system;
- a greater contribution to exposure rate from the apparent low-energy x rays in the chemical and volume control system than in the primary coolant system and components or in the refueling activities and spent fuel storage;
- a lesser contribution to exposure rate from low-energy x rays near steam generator Number 2 and in the shutdown cooling lines than in other areas;
- a greater contribution to exposure rate from Co-60 in the cavity drain lines and near steam generator Number 2 than in other areas of the plant;
- a lesser contribution to exposure rate from Co-60 in the spent fuel pool heat exchangers than in other areas of the plant;
- a greater contribution to exposure rate from Cs-137 in the shutdown cooling lines and in the pipe penetration room than in other areas of the plant;
- a lesser contribution to exposure rate from Cs-134 and Cs-137 in the cavity drain lines and letdown valve than in other areas of the plant.

An important aspect in evaluating the radionuclide contributions to exposure rates is to determine how they change as a function of time. An analyses of the changing source term can be made through a comparison of measurements

made during routine operations shortly before the outage and those repeated during the outage. Comparison of these gamma spectral measurements indicate that the radionuclide composition of the measured source terms changes significantly from the beginning to the end of the outage as a result of differences in the decay rates of individual radionuclides. Because the outage measurements were taken about eighty days following shutdown, the presence of a short-lived radionuclide with a decay half-life of less than approximately ten days would not be expected unless the radionuclide were a product of the decay of a longer-lived parent.

Calculations have been made of the estimated exposure rates at different times after reactor shutdown relative to the measurements made inside the primary pipe penetration room during routine operations. A summary of these calculations is presented in Table 4. Detailed calculations at different time periods (i.e., t = 0, 2, 5, 10, 20, 40, and 80 days) after plant shutdown are included. The contribution of radionuclides other than activation and fission products to the total exposure rate remained at low levels throughout this time period. Also included are data obtained in the pipe penetration room during the outage with the apparent low-energy photon (unknown) contribution eliminated so that direct measurement data could be compared to the calculated estimates. The exposure rates present during routine operations were measured several weeks prior to plant shutdown and were assumed to be the same as the exposure rates present at the time the plant was shutdown. The calculated exposure rates at the different times take into account only the radioactive decay of each radionuclide; no other parameters or assumptions have been included.

The data from outage time t=0 to t=80 days suggest that the estimated relative contributions of the activation products and the fission products to the total exposure rate appear to remain nearly constant over the time period studied. As the short-lived radionuclides decay to negligible levels, the longer-lived nuclides, such as Cs-137, Co-60, and Cs-134, dominate the radionuclides present and maintain a nearly constant ratio of activation product to fission product. The calculations of radionuclides present and their contribution to the exposure rates at 80 days after shutdown can be compared directly to the measurements obtained on approximately the eighty-first day of the outage. The apparent low-energy photon contribution was ignored for this comparison because the extrapolation of the data after plant shutdown was made based on radionuclides present and gamma peaks identified in the spectrum taken during routine operations measurement. The relative contributions of activation products and fission products to the total exposure rate compare favorably between the calculated estimates and the field observations. The total exposure rate calculated for the estimated and the measured values are different; the estimated exposure rate (3.0 mR/h) is greater than the measured exposure rate (1.7 mR/hr). This difference may result from the geometric correction factor, which was established through the analysis of the exposure rate present during routine operations and was kept constant for all calculations.

If the relative contributions of the activation products and fission products do remain constant during an outage period, any fuel cladding failure events would remain as significant to radiation exposure rates and occupational

radiation exposures during the outage as during routine operations. The significance of fuel cladding failures during routine operations has been documented in NUREG/CR-4485 Vol. 1 to be more than a five-fold increase in radiation exposure rates in specific areas of the plant.

Based on the data collected during the refueling and maintenance outage of an 1100 MWe PWR nuclear power plant that had been operating with fuel having fuel cladding failure characterized as 0.21 percent, the following conclusions can be drawn:

- The data indicate that the relative contributions of the activation products and the fission products to the total exposure rate is nearly constant over the duration of the outage. This constancy is due to the significant contribution to the exposure rates of the longer-lived isotopes of cesium (a fission product) and cobalt (an activation product).

- Because the relative contributions of the activation and fission products remain nearly constant over the duration of outage, fuel cladding failures are as significant a contributor to radiation exposure rates and occupational radiation exposure during the outage as they are during routine operations. As reported in NUREG/CR-4485 Vol. 1, fuel cladding failures increased radiation exposure rates an estimated 540 percent in some areas of the plant during routine operations. Consequently, it can be concluded that fuel cladding failures resulted in many areas of the plant having radiation exposure rates significantly greater during the outage than would have been present following routine operations under normal fuel conditions; that is, without failed fuel.

- The apparent low-energy photopeaks appeared to contribute significantly to exposure rates in most locations, except those near steam generator #2. Unfortunately, the exact cause of the low-energy photopeaks has not been determined conclusively. The origin of this photopeak should be determined and warrants further investigation. The uncertainty regarding the origin of the photopeak should not, however, confuse the comparison of the contribution from fission products versus activation products to radiation exposure rates nor the conclusions derived from this critical comparison.

- Evaluation of multi-element beta thermoluminescent dosimeter radiation depth-dose data indicate that beta radiation is not a significant contributor to radiation dose rates in those locations within the plant that were evaluated. This small contribution is perhaps due to the fact that much of the beta radiation was attenuated within the piping system.

- Analyses of air samples taken inside containment for radioactive particulates and radioiodine indicate that at eighty-two days into the outage airborne radioactive materials and gases were not a significant contributor to occupational exposures at the locations measured. Discussions with plant personnel indicated that airborne radioactive materials and gases may have been at sufficiently high concentrations to impact occupational exposures inside containment at the beginning of the outage if workers had not utilized respiratory protection.

- Data collected at locations associated with refueling activities and spent fuel storage indicate that Co-58 is a principal contributor to exposure rates in these locations of the plant. The primary means of producing Co-58 is by activation of Ni-58 through a neutron-proton reaction. The Co-58 may be present because nickel or inconel alloys are used in components or piping. Activation products such as Co-58 are transported from the primary system to the spent fuel storage area on the fuel assemblies.

CONTENTS

ABSTRACT iii

EXECUTIVE SUMMARY v

FIGURES xv

TABLES xvii

INTRODUCTION 1

METHODOLOGY 5

MEASUREMENT PROCEDURES 7

MEASUREMENTS 11

MEASUREMENT APPROACH 11

DATA COLLECTION 13

ANALYSIS AND DISCUSSION 27

RESULTS 29

CHEMICAL AND VOLUME CONTROL SYSTEM 31

Inside Primary Pipe Penetration Room (POM06) 31

Purifying Ion-exchanger Valve Room/Inlet (POM01) . . . 32

Bypass Ion-exchanger Valve Room/Outlet (POM04) . . . 32

Off-gas Line From Pressurizer Located in the Volume Control Tank Room - Plant Modification (POM02) . . . 32

Pressurizer Level Control Valve Room (POM3) 33

Charging Pump Room (POM05) 33

PRIMARY COOLANT SYSTEM AND COMPONENTS 34

Pressurizer Surge Line (POM16) 34

Hot Leg of Steam Generator #2 (POM13) 34

Valve Near Steam Generator #2 (POM08) 34

Letdown Valve Near Steam Generator #1 (POM10) . . . 35

Shutdown Cooling Line - 17-Foot 6-Inch Level (POM09) . . 35

Shutdown Cooling Line - 30-Foot Level (POM12) . . . 35

Shutdown Cooling Line Upstream of POM14 Valve (POM15) . . 36

Shutdown Cooling Isolation Valve (POM14) 36

REFUELING ACTIVITIES AND SPENT FUEL STORAGE 36

Refueling Cavity Drain Line Near Elevation Access Point (POM11) 37

Cavity Drain Line at Termination (POM07) 37

Spent Fuel Pool Pump Room (POM17) 37

Fuel Pool Heat Exchanger (POM18) 38

Fuel Pool Heat Exchanger - Main Line (POM19) 38

RADIONUCLIDE CONTRIBUTIONS 38

ESTIMATED EXPOSURE RATES AFTER REACTOR SHUTDOWN 40

CONCLUSIONS 45

BIBLIOGRAPHY 47

APPENDIX A - RADIONUCLIDE PRODUCTION A.1 48

APPENDIX B - GAMMA SPECTRAL ANALYSES B.1 52

APPENDIX C - ATTRIBUTABLE EXPOSURE RATES DERIVED FROM GAMMA SPECTRAL DATA C.1 72

APPENDIX D - ESTIMATED ATTRIBUTABLE EXPOSURE RATES AFTER REACTOR SHUTDOWN D.1 111

FIGURES

1	Equipment Arrangement for the Gamma Spectrometer		8
2	Efficiency Curve for a Ge Detector		10
3	Efficiency Curve for the Type of Ionization Chamber Used		10
4	Fuel Pool Cooling and Clean-up System		18
5	Typical PWR Systems Diagram		19
6	Auxiliary Building Floor Plan (9.0-foot elevation)		20
7	Auxiliary Building Floor Plan (24.0-foot elevation)		21
8	Auxiliary Building Floor Plan (50.0-foot elevation)		22
9	Containment Structure and Penetration Building Floor Plan (9.0-foot and 15.0-foot elevations)		23
10	Containment Structure Floor Plan (17.5-foot elevation)		24
11	Containment Structure and Penetration Building Floor Plan (30.0-foot elevation)		25
12	Gamma Spectra Obtained Inside Primary Pipe Penetration Room (GSM13) - During Routine Operations		41
13	Gamma Spectra Obtained Inside Pipe Penetration Room (POM06) - During an Outage		42
B.1	Gamma Spectra Obtained Inside Pipe Penetration (POM06)	B.2	53
B.2	Gamma Spectra Obtained at the Purifying Ion-exchanger Valve Room/Inlet (POM01)	B.3	54
B.3	Gamma Spectra Obtained Inside Bypass Ion-exchanger Valve Room/Outlet (POM04)	B.4	55
B.4	Gamma Spectra Obtained at the Pressurizer Off-gas Line (POM02)	B.5	56
B.5	Gamma Spectra Obtained Inside Pressurizer Level Control Valve Room (POM03)	B.6	57
B.6	Gamma Spectra Obtained Inside Charging Pump Room (POM05)	B.7	58
B.7	Gamma Spectra Obtained at the Pressurizer Surge Line (POM16)	B.8	59

B.8 Gamma Spectra Obtained at the Hot Leg of Steam Generator #2 (POM13) B.9 60

B.9 Gamma Spectra Obtained at the Valve Near Steam Generator #2 (POM08) B.10 61

B.10 Gamma Spectra Obtained at the Letdown Valve Near Steam Generator #1 (POM10) B.11 62

B.11 Gamma Spectra Obtained at the Shutdown Cooling Line - 17'6"-Level(POM09) B.12 63

B.12 Gamma Spectra Obtained at the Shutdown Cooling Line - 30'-Level (POM12) B.13 64

B.13 Gamma Spectra Obtained at the Shutdown Cooling Line Upstream of POM14 Valve (POM15) B.14 65

B.14 Gamma Spectra Obtained at the Shutdown Cooling Isolation Valve (POM14) B.15 66

B.15 Gamma Spectra Obtained at the Cavity Drain Line Near Elevation Access Point (POM11) B.16 67

B.16 Gamma Spectra Obtained at the Main Cavity Drain Line (POM07) B.17 68

B.17 Gamma Spectra Obtained Inside Spent Fuel Pool Pump Room (POM17) B.18 69

B.18 Gamma Spectra Obtained at the Fuel Pool Heat Exchanger (POM18) B.19 70

B.19 Gamma Spectra Obtained at the Fuel Pool Heat Exchanger - Main Line (POM19) B.20 71

TABLES

1	List of Measurement Equipment	7
2	Summary of Onsite Measurements	17
3	Summary of Onsite Measurement Results	30
4	Estimated Exposure Rates for Individual Radiation Sources and Percentage of Total Exposure Rate for Selected Times After Plant Shutdown	41
A.1	Activity of Fission Products in Curies at Specified Times (T) After Removal From a Reactor That Has Operated at 1000 kW Energy for 1 Year	A.2 49
A.2	Activation Products	A.3 50
A.3	Fission Products	A.4 51

BLANK PAGE

ACKNOWLEDGMENTS

The authors wish to acknowledge the efforts of Lyle A. Rathbun and Timothy P. Lynch in the collection of data instrumental to this study, and of Lyn E. Myers, Sallie J. Ortiz, David K. Hilliard, and Claudia H. Burk for their support contributions.

INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has the responsibility to monitor and regulate the operation of commercial U.S. nuclear power plants. When operating practices, safety issues, or regulatory goals indicate that a regulatory requirement may need to be changed, NRC considers both onsite and offsite radiation protection before making a decision. At present, there are a significant number of existing and potential generic issues involving regulatory decisions that focus on small incidents or abnormal events. Unlike events of a greater magnitude, such as severe core damage, small incidents are not generally expected to result in any increase in radiation exposure to the general public. Therefore, from a radiation protection standpoint, the radiological consequences of such events are limited primarily to their impact on the workers within the nuclear power plant. In order to resolve the numerous generic issues pertaining to small incidents, it is first necessary to quantify the significance of the potential onsite radiological consequences.

This report is the second in a series of case studies designed to evaluate the magnitude of increases in occupational radiation exposures at commercial U.S. nuclear power plants resulting from small abnormal events. The incident evaluated in each case study is fuel cladding failure. Although this event may have no measurable impact on the general public, it can result in elevated primary coolant activity and increased radiation exposure rates at some locations within a plant. This evaluation of the radiological impact of the fuel cladding failure events on nuclear power plant workers is intended to serve as an indicator of the significance of small incidents or abnormal events in general. The resulting information should be useful in the probabilistic risk assessment of the consequences of such events, and such assessments should, in turn, provide data to support the regulatory decision process.

Given that fuel cladding failures will be used as an indicator of the consequences of small incidents or abnormal events, it is important to determine when such failures have the potential to impact onsite radiological conditions. The magnitude of the impact of fuel cladding failure at a pressurized-water reactor (PWR) during routine operations has been quantified and reported in NUREG/CR-4485 Vol. 1 (PNL-5606 Vol. 1), "The Impact of Fuel Cladding Failure Events on Occupational Radiation Exposures at Nuclear Power Plants. Case Study: PWR During Routine Operations." For that study, measurements were taken in two mirror-image PWRs. One reactor was operating with fuel having excessive or abnormal cladding failures while the other was operating under normal fuel conditions.

With the evaluation of the impact of fuel cladding failure events on occupational radiation exposure for the PWR during routine operation completed, the next task is to evaluate the impact of failed fuel on occupational exposure rates during a subsequent maintenance and refueling outage. The purpose of this report is to document that specific task. To accomplish the task objectives, measurements were made during the subsequent maintenance and refueling outage at the PWR that had operated with fuel having excessive cladding failure. Measurements that had been taken in areas of routine worker

access while the plant was operating were repeated under the outage conditions and new measurements were taken inside the plant's containment building and spent fuel pool pump room.

An unexpected event that occurred at the PWR during the maintenance and refueling outage had a significant impact on the assessment of the consequences of fuel cladding failure events. As part of the refueling operation, fuel assemblies with leakers were to be replaced. Prior to being tested for leaking rods, the fuel assemblies were transferred to the spent fuel pool. During the testing process, a portion of a fuel rod identified as a leaker in one assembly was broken and a piece of it appeared to crumble. Small, highly-radioactive fuel particles were released into the pool. Subsequent maintenance and refueling activities resulted in some of these particles being picked up on tools and transported throughout the fuel handling building. Fortunately, from a radiation protection standpoint, the particles did not appear to become airborne for significant periods of time. Fuel particles were not observed in any air monitoring surveys or in any ventilation system filters during this study. The dispersal appeared to have been limited to the fuel handling building floor and locations of apparent worker contact such as hand-railings and door knobs. The cost associated with containing and removing these particles, and the occupational exposures resulting from their presence significantly impact the consequences of fuel cladding failure events.

In order to apply the results of this task to risk assessments of generic issues, it is necessary to consider the data within the context of the plant and fuel characteristics of the PWR having fuel with excessive cladding failure. When it entered the maintenance and refueling outage, the unit had been operating for approximately twenty-two months since its initial fuel load. Measurements were taken on the eighty-first and the eighty-second days of the outage. The PWR is power rated as a 3560 megawatt-thermal (1100 megawatt-electric [MWe]) unit. When NUREG/CR-4485 Vol. 1 was completed in November 1985, the best available estimate of the fuel cladding failure events characterized the degraded fuel as 62 pin-hole leakers, which translates to 0.13 percent failed fuel. This estimate suggests that the level of degraded or failed fuel was more than ten times that of the normal fuel operating conditions, although that level is still within the plant's technical specifications. As a result of detailed inspection and testing of individual assemblies of the core in December 1985 and January 1986, the final estimate on the number of leaking fuel rods was 105. These rods were contained in 12 fuel assemblies. Considering that there are a total of 49580 fuel rods within the reactor core, the 105 pin-hole leakers translate to 0.21 percent failed fuel.

There is another plant characteristic affecting the application of these data to risk assessment. As a result of elevated primary coolant activity, the pressurizer steam space sample line in the PWR with failed fuel was modified to provide for continuous venting of the steam space. This change was made to enhance the removal of fission product gases from the reactor coolant system. The impact of this modification during operation was to reduce the equilibrium coolant activity levels by an estimated 30 to 40 percent. When considering how the data from this study could be applied to a PWR that does not have the pressurizer steam space vent (off-gas) line, it is

necessary to determine which plant locations, systems and components would experience greater radiation exposure rates at the beginning of the outage as a result of higher coolant activity and radiation levels from the steam space effluent.

Other sections of this report discuss the methodology and theory of the radiation measurement techniques used, the location and equipment setup for each measurement, analysis of the results, and suggested conclusions to be drawn from the data. Included in the appendices are a review of the principal means of production for many of the radionuclides of interest, a presentation of the gamma spectral analyses for all of the measurements, and a computer analysis and estimation of the radiation exposure rate attributable to each radionuclide identified in each gamma spectrum.

BLANK PAGE

METHODOLOGY

The primary objective of this task was to collect radiation measurement data to determine the contribution of fuel cladding failures on occupational radiation exposures during a PWR maintenance and refueling outage. To meet this objective, a study design is presented that complements previously collected data and attempts to minimize the many potentially confounding variables. Key aspects of the study design include 1) the selection of the nuclear power plant site where the radiation measurements were to be collected, 2) the locations where the measurements were to be made, 3) the types of measurements to be performed, and 4) the supporting data to be requested from plant personnel.

For the previous study of radiation exposure rates during routine operations, a PWR was identified that was operating with an excessive number of fuel cladding failures. Because the schedule for the unit included an upcoming maintenance and refueling outage, it was prudent to use the same unit for this task. The PWR where the measurements were taken is referenced as Unit B in NUREG/CR-4485 Vol. 1.

The identification of appropriate measurement locations was based on several factors including measurement locations used for the PWR "During Routine Operations" task, areas of worker activity during the maintenance and refueling outage, areas near major system components within containment, areas throughout the primary coolant system loop, and areas within the spent fuel pool pump room. Using the locations where measurements were taken while the unit was operating facilitated both the comparison of the routine operations gamma spectra with the outage gamma spectra and the analysis of the radionuclide compositions. In addition, this approach provided an indication of the changing source term with the decay of each radionuclide. Major system components of interest within containment included the pressurizer and steam generators.

Measurements were performed using four basic techniques. First, an intrinsic germanium (Ge) detector was used to distinguish the energy of the gamma photons through spectral analyses. Second, at the same time and location of the gamma measurements, exposure rates were determined using a portable ionization chamber. Third, at some gamma measurement locations where significant beta contributions were expected, eight-element thermoluminescent dosimeters (TLDs) were exposed for a timed interval. Finally, near measurement locations where airborne radionuclide concentrations were suspected, air samples were collected with a high-efficiency filter and silver-zeolite cartridge for analyses of particulates and iodine, respectively.

As a result of these measurements, it was possible to determine the percentage of the exposure rate at each measurement location attributable to each radionuclide identified through the gamma spectrum analyses. The categories of radionuclides included fission products, activation products, and other means of gamma photon and x-ray production including those radionuclides occurring naturally. The ability to distinguish the radiation exposure contribution of fuel cladding failure from that of corrosion product buildup and other sources was the key to determining whether fuel cladding failure events

significantly impact the exposure rates in the plant. The TLD measurements were designed to quantify the beta contribution to the exposure rates, and the air samples were used to estimate the potential contribution to occupational exposure via inhalation.

MEASUREMENT PROCEDURES

The measurement procedures used for this task were similar to those used during the collection of data for NUREG/CR-4485 Vol. 1 with the addition of the multi-element beta TLDs. Four different types of measurements were taken. These measurements provided gamma spectral data, radiation exposure rates, airborne radionuclide concentrations, and TLD radiation depth-dose data. The following section discusses the equipment and procedures used for each type of measurement. An equipment list for each type of measurement performed is presented in Table 1.

TABLE 1. List of Measurement Equipment[1]

EQUIPMENT DESCRIPTION	Model Number	Serial Number
1. Photon Spectroscopy		
Ortec Intrinsic Germanium Detector and Preamplifier	GEM10175	1081426
Canberra Series 40 Multichannel Analyzer including Analog to Digital Converter, Linear Amplifier, and Power Supply	4203	25-P-76RB
Canberra Cassette Interface	5421F	6841847
Realistic (Miniset 9) Cassette Recorder	5421M	2831106
2. Exposure Rate Measurements		
Eberline Air Ion Chamber	RO-2	2530
3. Air Sampling		
RadeCo Hi-Volume Portable Sampler	H-809B-1	3987
Open-Face Combination Filter and Cartridge Holder	47mm/SAI/809	None

[1] The identification of equipment used in this study is not intended as an endorsement or recommendation for their use.

Gamma spectral measurements were made to identify the contributing radionuclides. The system included an intrinsic Ge detector, a preamplifier, a high-voltage power source, an amplifier, and a multichannel analyzer. Data on the collected spectra were stored on magnetic tape cassettes for transfer to a computer for analyses. The multichannel analyzer was calibrated to 1 keV per channel over 4096 channels. The equipment block diagram is presented in Figure 1. Equipment calibration was verified before the measurements were taken.

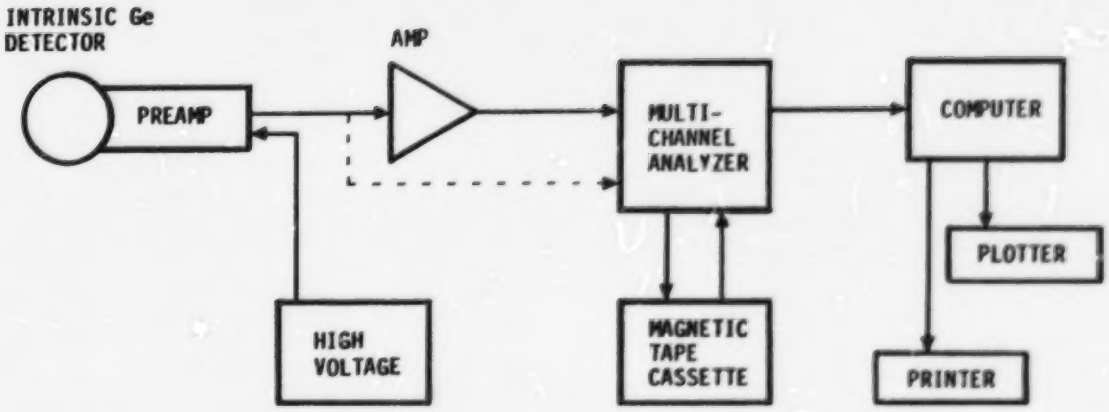


FIGURE 1. Equipment Arrangement for the Gamma Spectrometer

The magnitude of the photon flux is important when taking gamma spectral measurements. For measurements taken during routine operations as documented in NUREG/CR-4485 Vol. 1, a lead collimator was used to reduce the number of photons incident on the detector when the exposure rate would have otherwise been too high. For measurements taken during the outage, a lead collimator was not used. Rather, the detector-to-primary-source distance was increased when the exposure rate was greater than approximately 10 mR/h at the detector location. This change did not preclude repeating any of the routine operations measurements in the same detector setup because the exposure rates in the previous locations did not approach 10 mR/h at the time of the outage measurements.

Raw spectral data were collected as pulse-height distributions. These distributions were corrected for detector efficiency and scattering losses during analyses. A computer code was used to determine which spectral peaks were statistically significant and the number of counts above background associated with each peak. Distinct peaks not identified by the code were processed by a manual method. An energy calibration performed prior to the measurements allowed the radionuclide responsible for each photon peak to be identified by its characteristic energy.

At each measurement location, a portable ionization chamber survey instrument was used to determine the exposure rate at the intrinsic Ge detector position. In each case, the survey reading was taken with the instrument held in the same physical location as the detector. The instrument used was selected from instruments routinely used at the plant. Calibration of the instrument was verified by checking against calibration records.

This type of ionization chamber is typically used to measure beta, gamma, and x-ray radiation, with four linear ranges of operation from 5 to 5,000 mR/h. The detector is air filled, vented to the atmosphere, and fitted with a

mylar window and a sliding aluminum beta shield. For this task, all readings were made with the beta shield closed. The window was closed to maintain electron equilibrium for higher-energy photons. In addition, the closed window measurement corresponded more favorably with the aluminum end cap on the intrinsic Ge detector.

Analyses of the data included consideration of the efficiency of the intrinsic Ge detector and the portable ionization chamber. The efficiency of a detector as a function of photon energy can be determined through measurement of sources with known activities. The efficiency curve for an intrinsic Ge detector similar to that used for the gamma photon spectroscopy is presented in Figure 2. The low-energy decline in the efficiency is due to the thickness of the detector window, while the high-energy decline is due to the increased penetration power of higher energy photons. The efficiency curve for a portable ionization chamber similar to that used for the exposure rate measurements is presented in Figure 3. The ion chamber is essentially linear above 40 keV with the sliding aluminum beta shield closed. The low-energy decline in the efficiency is due to the thickness of the detector window.

Air samples were taken at locations with airborne concentrations thought to be representative of those inside containment. Air samples were drawn using a high-volume portable air sampler with a combination filter and cartridge holder. The sampler and holder were selected from instruments routinely used for plant surveys. The air sampler provided a continuously adjustable flow rate from 0 to 8 cubic feet per minute (cfm) and the holder assembly accommodated a 2-inch-diameter filter paper and a 2.25-inch-diameter cartridge. The particulate filter paper used was designed to provide a high collection efficiency for approximately 0.3-μm-diameter or larger particles at the flow rates created by the sampler. The cartridges used were silver-impregnated zeolite cartridges that were 37 percent silver by weight. They have an essentially 100 percent collection efficiency for iodine. The samples were drawn using the plant's collection procedure, which requires a 4-cfm flow rate for 10 minutes, thus yielding a sample volume of 40 cubic feet. All filters and cartridges were counted using the plant's lithium-drifted germanium spectroscopy system and analyzed using peak identification software.

At some measurement locations, a multi-element beta TLD was placed and exposed to the radiation field in order to evaluate the beta depth-dose. Each TLD was comprised of an eight-element dosimeter with various amounts of filtration over each element. An indication of the beta energy spectrum of the source can be obtained by comparing the relative response of the elements under the different filters. Each element contains three TLD chips with mylar or aluminum filtration. The thicknesses of the seven aluminum filters used were 1, 5, 10, 20, 32, 64, and 125 mils (one mil is equal to one one-thousandth of an inch). These values correspond to density thicknesses of 6.86, 34.3, 68.6, 137, 219, 439, and 857 grams per square centimeter (g/cm^2), respectively. The aluminized mylar filter on the other element had a thickness of 0.25 mils.

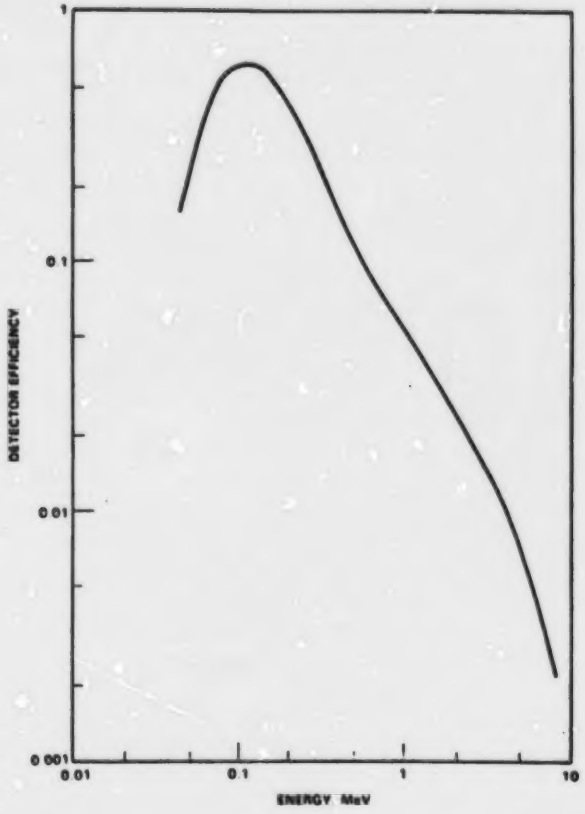


FIGURE 2. Efficiency Curve for a Ge Detector

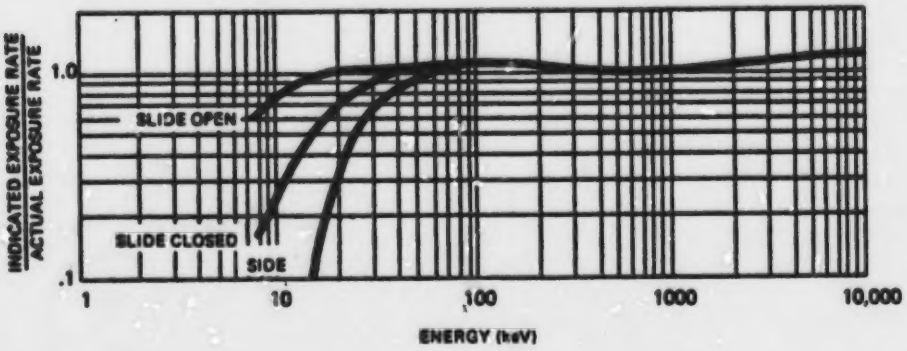


FIGURE 3. Efficiency Curve for the Type of Ionization Chamber Used

BEST COPY AVAILABLE

MEASUREMENTS

To accomplish the task objectives, the measurement procedures described in the preceding section were used to collect data in selected locations of a PWR during a maintenance and refueling outage. As previously described, the measurements were made at the same two-unit PWR plant site as reported in NUREG/CR-4485 Vol. 1. For this task, however, measurements were made only in the unit (Unit B) that had operated with fuel having excessive cladding failures.

The two-unit PWR plant site is comprised of two mirror-image containment buildings with adjacent pipe penetration buildings separated by a common auxiliary building that contains the radwaste as well as the chemical and volume control systems. The unit with degraded fuel had been operating at 55 percent of the licensed maximum core power for approximately 22 months when the unit was shut down for its maintenance and refueling outage. The extent of fuel damage has been characterized as 105 pin-hole leakers in 12 assemblies, which is equivalent to 0.21 percent failed fuel. Measurements were taken on the eighty-first and eighty-second days of the scheduled outage.

The remainder of this section provides details of the measurement approach and the locations of specific measurements.

MEASUREMENT APPROACH

In the previous study, measurements were taken in both units of the plant to allow for comparison of exposure rates between a unit with failed fuel (Unit B) and a unit without failed fuel (Unit A) during routine operations. For this task, it was recognized that measurements needed to be taken in Unit B during its subsequent maintenance and refueling outage. Based on the plant schedule, it was not prudent to wait for Unit A to enter its maintenance and refueling outage to obtain comparative measurements. Therefore, all measurements were taken in Unit B only.

Measurement locations were selected in Unit B based on the need to evaluate the impact of fuel cladding failures on occupational radiation exposures. Within this framework, measurement locations were selected on the basis of radiation exposure rates and required worker access during the outage. Nineteen measurements were taken at nineteen different locations within the unit. Because much of the outage work was performed inside containment, ten of the measurements were taken at selected locations within the containment structure. Of the remaining nine measurements, five were taken in the radwaste and chemical and volume control areas of the auxiliary building. The remaining four measurements were taken in the pipe penetration building which is located between the containment building and the auxiliary building side.

Information on the measurements and plant systems is presented in tables and figures located at the end of this section of the report. Table 2 presents a summary of information for each of the measurements, including a brief description of the detector orientation, physical characteristics and location. Figure 4, a schematic of the fuel pool cooling and clean-up system and

Figure 5, a schematic of the major systems within a typical PWR, illustrate the relative locations where measurements were made. Figures 6 through 11 are floor plans that identify the measurement locations on each of the plant elevations. Because of severe contamination problems experienced as a result of the broken fuel rod, no measurements were taken in the spent fuel handling building and spent fuel pool area. Measurements were taken, however, in the spent fuel pool pump room and spent fuel heat-exchanger room.

In order to determine the primary contributors to exposure rates, measurements were taken at strategic locations in the piping systems of both the chemical and volume control system and the primary reactor coolant system. Although these systems were not in service during the outage, surveys confirmed that the primary radiation sources in the plant buildings were radionuclides trapped within the components and piping comprising these systems. Examination of radiation survey maps and discussions with plant health physicists and engineers provided information to correlate specific areas of outage activity with significant exposure rates. With the aid of plant personnel, the primary radiation source in each area selected for measurement was identified.

To facilitate a limited comparison, selected measurements taken in Unit B during routine operations were repeated during the outage. These included five measurements in the auxiliary building and one in the pipe penetration building. In each case, care was taken to duplicate the original measurement setup as precisely as possible. During routine operations, high radiation exposure rates and elevated concentrations of airborne contamination had prohibited taking measurements inside the containment building. Consequently, there were no previous containment measurements for comparison. During routine operations, radiation levels in the spent fuel pool and associated spent fuel pool pumps, heat-exchangers, and recirculation piping had not been affected by the fuel cladding failure. However, it was determined that exposure rates in these areas had increased as a result of the refueling operation. Data on these increases were thus significant in quantifying the impact of the cladding failure on occupational exposures. As was the case with the containment building measurements, there was no previous information to compare with the measurements taken near the spent fuel pool systems and recirculation piping.

When positioning the intrinsic Ge detector to take a measurement, consideration was given to reproducibility as well as to the location of primary and secondary sources relative to the detector. In most instances, measurements were taken with the detector supported at a distance of 39 in. from the floor. In the case of an elevated primary radiation source such as an overhead pipe, the detector was positioned to facilitate the best characterization of the primary radiation source and to minimize interference from secondary sources. Frequently, this entailed raising the detector level in order to position it closer to the primary source.

When considering the detector orientation and location, the magnitude of the exposure rate at the detector was also important. If the radiation exposure rate from the primary source, as measured at the detector with the portable ionization chamber, was greater than approximately 10 mR/h, the detector-to-primary-source distance was increased. An exposure rate in excess

of 10 mR/h proved to be a good indicator that the electronic dead time of the detector may significantly affect the accuracy of the measurement. In those instances when the source-to-detector distance was increased, care was taken to avoid introducing secondary radiation sources.

Thermoluminescent dosimeter exposure and air sample measurements were taken at or near selected locations used for gamma spectral analyses. Multi-element beta TLDs were also placed and exposed for discrete time periods at several measurement locations to verify the exposure rates measured with the ionization chamber and to determine if a significant beta radiation component was potentially contributing to occupational exposure. Air samples were taken at selected locations to determine the extent of airborne radionuclide contributions to occupational exposure. Table 2 summarizes the orientation and location information for these measurements.

DATA COLLECTION

As may be noted in Table 2, measurement Number 1 was taken in the cubicle containing the purifying ion-exchanger valves. This cubicle is located on the 50-foot elevation and dimensions of approximately 4 feet by 10 feet. The detector was placed just inside the entrance at a distance of 46.5 in. from the pipe containing the control valve. The detector was supported 39 in. from the floor and was oriented towards the pipe. For this measurement, the gamma spectral collection time was 300 live time seconds.

Measurement Number 2 was made at the 50-foot elevation in the cubicle housing the valves for the pressurizer off-gas line. This line runs from the steam space of the Unit B pressurizer to the volume control tank. As previously described, the pressurizer steam space sample line had been modified to provide for continuous venting of the steam space to enhance the removal of fission product gases. The modification was successful in reducing primary coolant activity but created new radiation concerns. The cubicle housing the valves measures approximately 11 feet by 6 feet, and the detector was positioned just inside the entrance and beyond the baffle wall. The detector was supported 39 in. from the floor and was oriented toward the off-gas line at a distance of 68 in. For this measurement, the gamma spectral collection time was 300 live time seconds.

Measurement Number 3 was taken in the boric acid pressure-level control valve room. This measurement was a repeat of a measurement taken during routine operation as documented in NUREG/CR-4485 Vol. 1. Since publication of that document, it has been determined that the boric acid make-up system is a once-through process and that it is not possible for any radioactive material to collect in the boric acid pressure-level control valve. It has also been determined that the radiation detected and measured in this room is the result of contamination of the pressurizer-level control valve. The pressurizer make-up system is not a once-through process and this feature contributed to the apparent source of radiation exposure in this room. The cubicle containing the pressurizer-level control valve is located at the 24-foot elevation and has dimensions of approximately four feet by four feet. For this measurement, the detector was located just inside the entrance, supported 39 in. from the floor, and placed 37 in. from the pipe containing the control valve. For this measurement, the gamma spectral collection time was 300 live time seconds.

Measurement Number 4 was taken inside the cubicle housing the bypass ion-exchanger valves. This cubicle is located at the 24-foot elevation and measures four feet by four feet. The detector was positioned just inside the entrance and supported 39 in. from the floor. It was aimed directly at the middle of the pipe containing the control valve at a distance of 27 in. For this measurement, the gamma spectral collection time was 300 live time seconds.

The charging pumps are located on the 9-foot elevation and are housed in concrete rooms measuring approximately 12 feet by 20 feet. Measurement Number 5 was taken in one room with the detector positioned just past the wall separating the valve room from the pump room and oriented towards the wall supporting the inlet and outlet piping. The detector was supported 39 in. from the floor and positioned approximately 112 in. from the pipes. For this measurement, the gamma spectral collection time was 600 live time seconds.

Measurement Number 6 was made on the 30-foot level of the penetration building inside the room containing the piping penetrations for the chemical and volume control system. The room is approximately 20 feet by 50 feet, one wall of which is the curved wall of the containment structure. The detector was supported 39 in. from the floor, placed just inside the room beyond the baffle wall and oriented towards the primary piping lines. For this measurement, the gamma spectral collection time was 300 live time seconds.

With one exception, all remaining measurements were made with a gamma spectral collection time of 100 live time seconds. Measurement number 18 was made with a gamma spectral collection time of 300 live time seconds.

Measurements Number 7 through 11 were taken on the 17.5-foot level of the containment structure. This is the lowest level of the containment building in which routine and outage activities are normally conducted. Measurements 7 and 11 were taken near two of the cavity drain lines. These lines had been identified as areas of high exposure rates due to the accumulation of radionuclides during cavity draining activities associated with the outage.

Measurement Number 11 was taken at the cavity drain line located on the side of the refueling cavity liner housing nearest the elevator shaft. Due to the nearness of the elevator access, this location was a high-traffic area during outage operations. The measurement was taken with the detector supported 39 in. from the floor and 21 inches from the drain line. Measurement Number 7 was taken at the main cavity drain line located on the opposite side of the refueling cavity liner housing from the other drain line. Although not located in such a heavily traversed area, the main line was a more significant radiation source necessitating the positioning of the detector approximately 30 feet from the drain line. As in previous cases, the detector was supported 39 in. from the floor. A multi-element beta TLD was also placed in this location near the position of the detector.

Measurement Number 8 was taken between the steam generator Number 2 enclosure area and the emergency drain sump at containment elevation 17.5 feet. The radiation source was valve HV9433 and the associated piping. Because the primary source was in a radiation zone, the exact source-to- detector distance

could not be measured. It was estimated that the detector was located 12 feet from the valve and supported 39 in. from the floor.

Measurement Number 10 was taken on the letdown valve located just inside the steam generator Number 1 enclosure area at the 17.5-foot level of the containment structure. The detector was located 9.9 feet from the valve and supported 39 in. from the floor. The valve was positioned in a congested corridor that required one to pass in close proximity to the valve.

Discussions with plant personnel and a review of survey results revealed that the shut-down cooling line was considered a significant source of radiation exposure within containment. Consequently, four measurements on two plant elevations were made on this line. Measurement Number 9 was taken at the 17.5-foot elevation on the section of shut-down cooling line adjacent to the containment outer wall near steam generator Number 1. This area serves as a primary corridor around the outer edge of containment. The detector height was 45 in. and the detector was located 52 in. from the pipe. A multi-element beta TLD was placed near this location. During its exposure, the TLD was positioned 83 in. from the floor and 40 in. from the pipe. The next three measurements on the shut-down cooling line were all taken on the 30-foot elevation. Measurement Number 12 was taken on the shut-down cooling line just outside the pressurizer enclosure area where the pipe passes to the next lower elevation. This area is near the outer corridor walkway and is on the same elevation as the equipment hatch. The detector was placed 94 in. from a radioactive hot spot on the pipe. A multi-element beta TLD was also placed near this location at a distance of 24 in. from the hot spot.

Measurements Number 14 and 15 were also made at the 30-foot elevation inside containment. The measurements focused on two locations in the steam generator Number 1 room. In this room, the shutdown cooling line runs overhead near the walkway around the steam generator. Measurement Number 14 was taken near the shutdown cooling line isolation valve positioned near the entrance to the enclosure. The detector was placed approximately 10 feet from the valve and supported 39 in. from the floor. The detector was oriented towards a radioactive hot spot near the valve, which was labeled as having an exposure rate of 150 mR/h on contact. In addition to its proximity to the walkway, another reason for taking a measurement at the valve was to attempt to compare the radionuclides held in a suspected crud trap near the valve with those upstream of the hot spot. The identification of the radionuclides causing the hot spot could be accomplished by comparing the relative contribution among radionuclides in the crud trap (Measurement 14) with those upstream of the hot spot (Measurement 15). Measurement Number 15 was taken for comparison on the same pipe upstream of the shutdown cooling line isolation valve and in the middle of a long stretch of straight piping. The detector was placed 88 in. from the floor and 21 in. from the overhead pipe.

Measurement Number 13 was made on the 30-foot elevation inside containment on the hot leg from steam generator Number 2. The detector was 48 in. from the hot leg and 11 in. from the steam generator and was supported 39 in. from the floor. A multi-element beta TLD was exposed two in. from the hot leg piping near this location.

The pressurizer surge line on the 30-foot elevation was the focus of the last measurement made inside containment. This line is located inside the pressurizer enclosure area and is approximately 20 feet above the floor. The detector was placed 14 feet from the floor and approximately six feet from a radioactive hot spot on the pipe labelled by plant personnel as having an exposure rate of 308 mR/h on contact. A multi-element beta TLD was exposed at the same location as the detector.

The next three measurements were taken in the pipe penetration building on systems associated with the spent fuel pool. Due to the rupture of the fuel rod that occurred during the refueling operation, the spent fuel pool area and associated systems had become highly contaminated. As a result, it was not possible to enter these areas to take measurements without potentially contaminating the measuring equipment. Consequently, it was not possible to obtain data directly from the fuel pool area. However, measurements were made in adjacent accessible areas (see below) in an effort to characterize the radionuclides contributing to exposure rates within this system.

One such measurement (Number 17) was taken at the 15-foot level of the penetration building inside the spent fuel pool pump room. The detector was positioned 24.5 in. from the inlet and outlet pipes of the pump and supported 39 in. from the floor. Measurement Number 18 was taken in the fuel pool heat exchanger room on the 30-foot elevation with the detector located 12 inches from the cold leg of the heat-exchanger assembly supported 39 in. from the floor. As previously reported, the gamma spectral collection time for this measurement was 300 live time seconds. Measurement Number 19 was taken in the hallway outside of the spent fuel pool heat-exchanger room. The main feed lines to the heat-exchanger were located in this hallway. Two pipes were found to be contributing to exposure rates in this area. One was hung from the ceiling above the hallway and the other from the hallway wall. The detector was positioned 57 in. from the ceiling pipe and 20 in. from the pipe on the wall while being supported 39 in. above the floor.

Discussions with health physics personnel and a review of the routine air sample monitoring results revealed that neither the auxiliary, penetration, nor containment building had been experiencing any airborne contamination problems during the outage. Since the containment building was considered the primary location where airborne contamination could be at detectable concentrations, air samples were drawn at different containment building elevations. The first was drawn on the 45-foot level outside the steam generator Number 1 enclosure area. The second was taken on the 17.5-foot elevation near the emergency drain sump area. The results from both measurements confirmed the absence of detectable airborne radionuclide contamination.

TABLE 2. Summary of Measurements

PWR Outage Measurement Number	Gamma Spectrometer Detector Height (in)	Distance From Primary Source (in)	Gamma Spectral Collection Time (Live Time Sec)	Exposure Rate at Detector (mR/h)	Exposure Rate on Contact/Primary Source (mR/h)	Multi-element Beta/TLD Exposed*	Measurement Location Building	Building Elevation (Feet)	Measurement Area Description
1	39	46.5	300	0.3	10	N	Rad. Waste	50	Purifying Ion-exchanger Valve Room/Inlet
2	39	68	300	0.2	3	N	Rad. Waste	50	Pressurizer Off-gas Line
3	39	37	300	1.5	8	N	Rad. Waste	24	Pressurizer Level Control Valve Room
4	39	27	300	2.0	18	N	Rad. Waste	24	Bypass Ion-exchanger Valve Room/Outlet
5	39	112	600	0.1	2.9	N	Rad. Waste	9	Charging Pump Room
6	39	N/A*	300	1.7	N/A*	N	Penetration	30	Inside Pipe Penetration Room
7	39	360	100	4.0	---	Y	Containment	17.5	Cavity Drain Line At Termination
8	39	144	100	6.0	45	N	Containment	17.5	Valve Near Steam Generator #2
9	45	52	100	6.5	45	Y	Containment	17.5	Shutdown Cooling Line - 17'6"-Level
10	39	119	100	5.0	---	N	Containment	17.5	Letdown Valve Near Steam Generator #1
11	39	21	100	4.0	6	N	Containment	17.5	Cavity Drain Line Near Elevation Access Point
12	39	94	100	4.1	800	Y	Containment	30	Shutdown Cooling Line - 30'-Level
13	39	48	100	5.0	5	Y	Containment	30	Hot Leg of Steam Generator #2
14	39	120	100	6.0	150	N	Containment	30	Shutdown Cooling Isolation Valve
15	88	21	100	4.2	7	N	Containment	30	Shutdown Cooling Line Upstream of POM14 Valve
16	168	72	100	8.0	308	Y	Containment	30	Pressurizer Surge Line
17	39	24.5	100	3.0	8	N	Penetration	15	Spent Fuel Pool Pump Room
18	39	12	300	2.6	4.2	N	Penetration	30	Fuel Pool Heat Exchanger
19	39	57	100	7.0	N/A**	N	Penetration	30	Fuel Pool Heat Exchanger - Main Line

* N = No; Y = Yes

** More than one primary source.

BEST COPY AVAILABLE

17

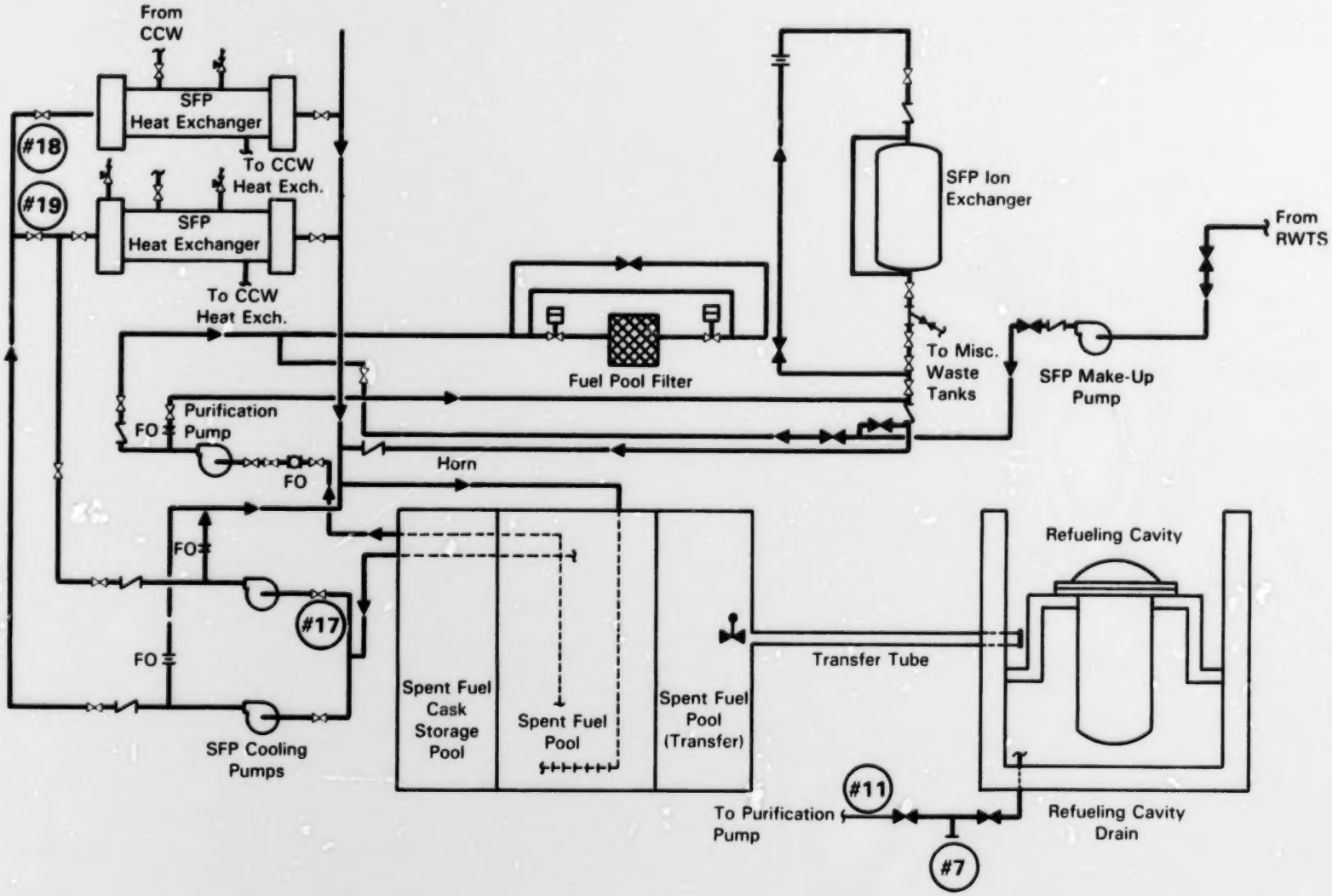
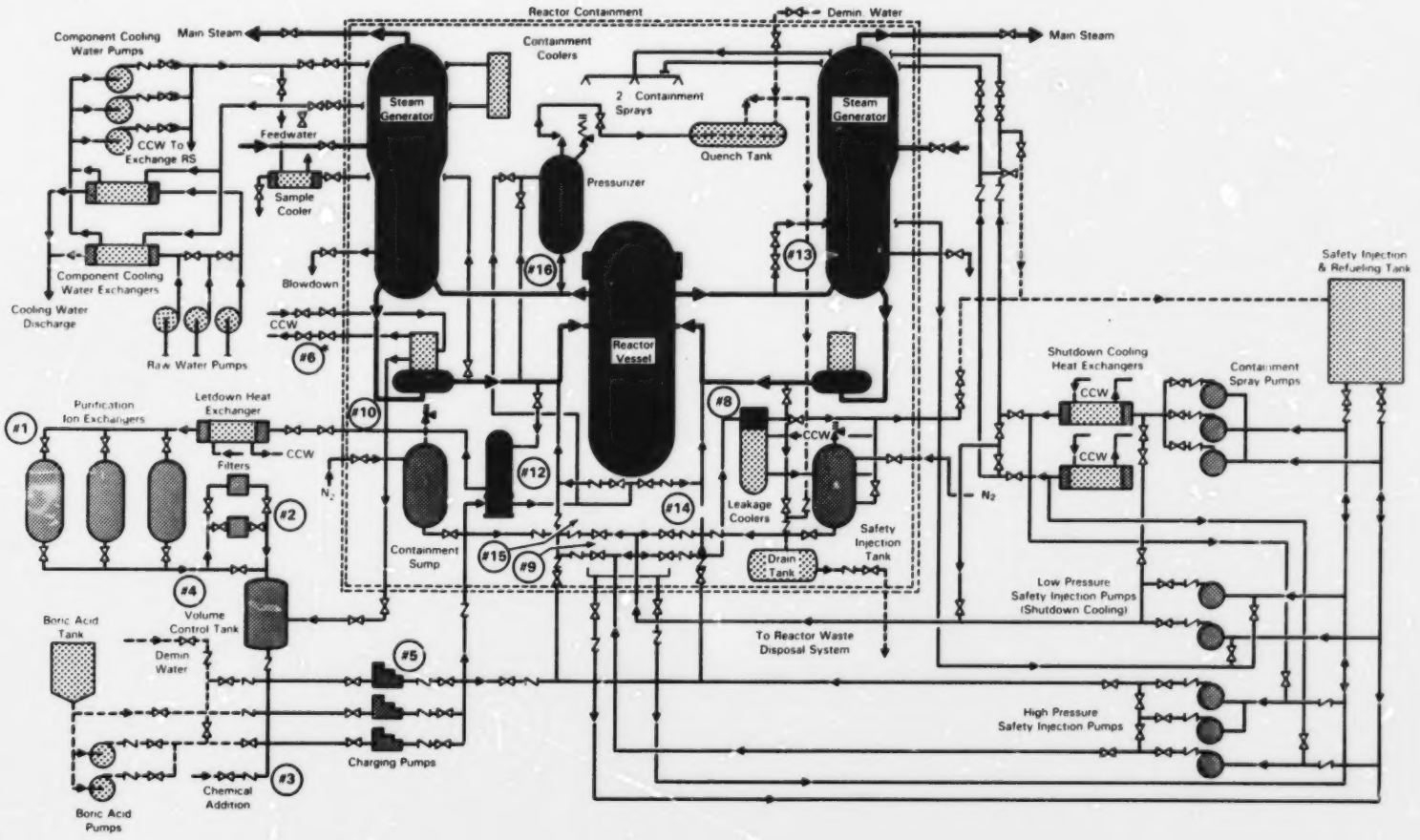


FIGURE 4. Fuel Pool Cooling and Clean-up System (Measurement locations Number 7, 11, 17, 18, and 19 are indicated).

BEST COPY AVAILABLE

18



* Penetration area measurement

FIGURE 5. Typical PWR Systems Diagram (showing measurement locations Number 2, 3, 9, 10, 12, 13, 14, 15, and 16).

BEST COPY AVAILABLE

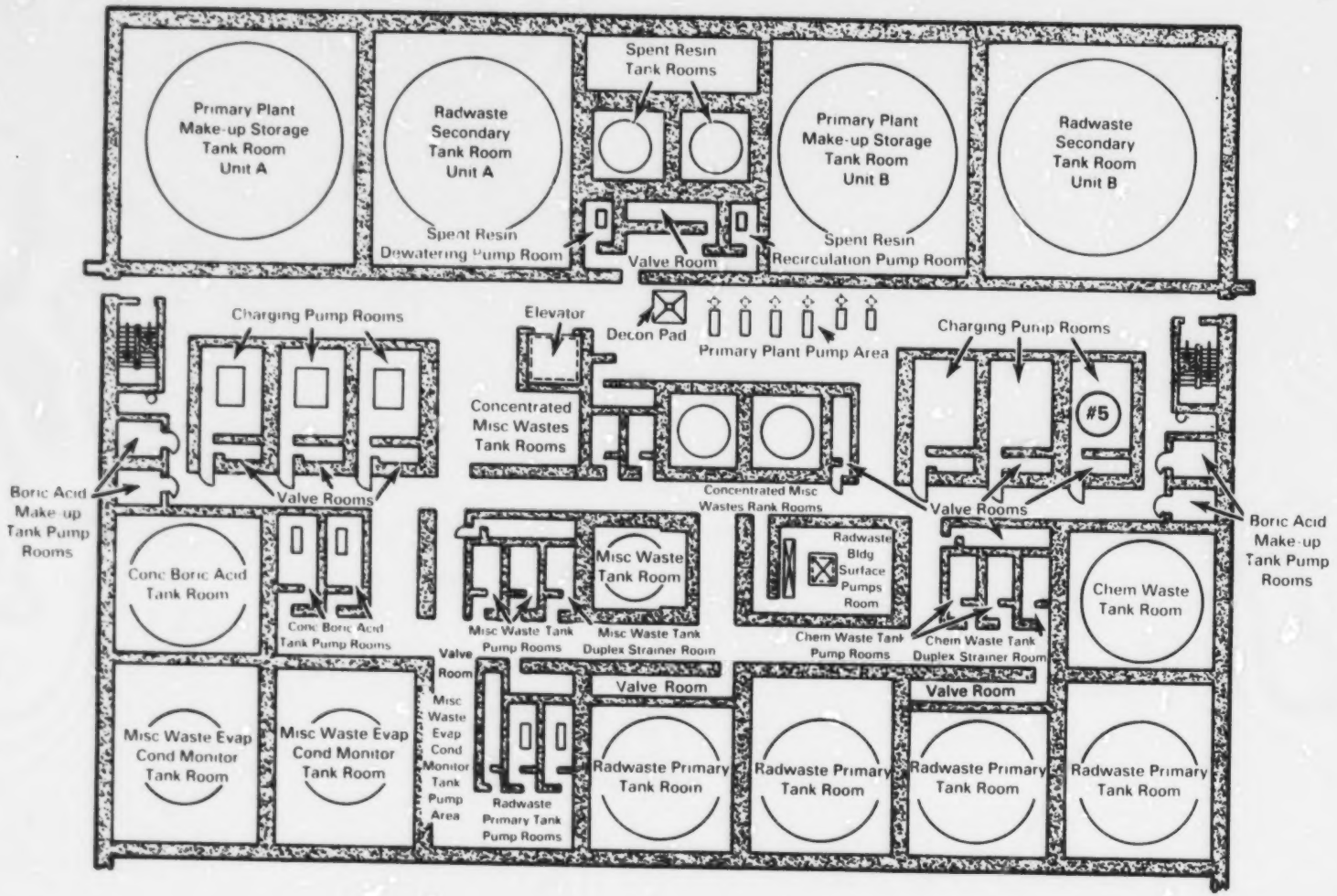

Primary Plant Make-up Storage Tank Room Unit A
Radwaste Secondary Tank Room Unit A
Spent Resin Tank Rooms
Primary Plant Make-up Storage Tank Room Unit B
Radwaste Secondary Tank Room Unit B
Spent Resin Dewatering Pump Room
Valve Room
Spent Resin Recirculation Pump Room
Charging Pump Rooms
Elevator
Decon Pad
Primary Plant Pump Area
Charging Pump Rooms
#5
Concentrated Misc Wastes Tank Rooms
Boric Acid Make-up Tank Pump Rooms
Valve Rooms
Concentrated Misc Wastes Rank Rooms
Valve Rooms
Boric Acid Make-up Tank Pump Rooms
Conc Boric Acid Tank Room
Conc Boric Acid Tank Pump Rooms
Misc Waste Tank Room
Misc Waste Tank Pump Rooms
Misc Waste Tank Duplex Strainer Room
Radwaste Bldg Surface Pumps Room
Chem Waste Tank Pump Rooms
Chem Waste Tank Duplex Strainer Room
Chem Waste Tank Room
Valve Room
Misc Waste Evap Cond Monitor Tank Room
Misc Waste Evap Cond Monitor Tank Room
Misc Waste Evap Cond Monitor Tank Pump Area
Radwaste Primary Tank Pump Rooms
Valve Room
Radwaste Primary Tank Room
Radwaste Primary Tank Room
Valve Room
Radwaste Primary Tank Room
Radwaste Primary Tank Room

FIGURE 6. Auxiliary Building Floor Plan (9.0-ft elevation) (showing measurement location Number 5).

BEST COPY AVAILABLE

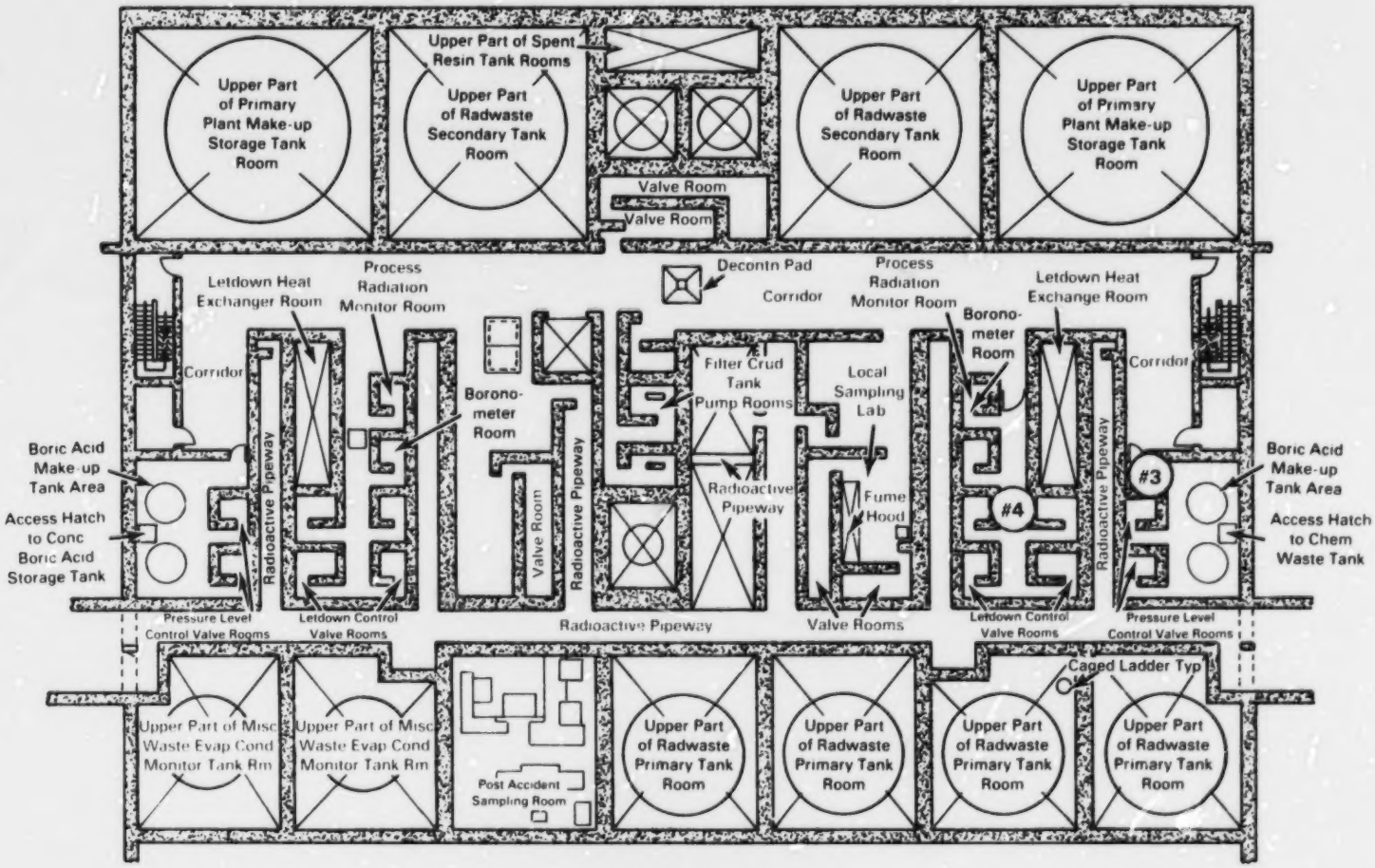

Upper Part of Primary Plant Make-up Storage Tank Room
Upper Part of Spent Resin Tank Rooms
Upper Part of Radwaste Secondary Tank Room
Valve Room
Valve Room
Upper Part of Radwaste Secondary Tank Room
Upper Part of Primary Plant Make-up Storage Tank Room
Letdown Heat Exchanger Room
Process Radiation Monitor Room
Decontn Pad
Corridor
Process Radiation Monitor Room
Borono-meter Room
Letdown Heat Exchange Room
Corridor
Corridor
Filter Crud Tank Pump Rooms
Local Sampling Lab
Borono-meter Room
Radioactive Pipeway
Boric Acid Make-up Tank Area
Access Hatch to Conc Boric Acid Storage Tank
Radioactive Pipeway
Valve Room
Radioactive Pipeway
Radioactive Pipeway
Fume Hood
#4
#3
Radioactive Pipeway
Boric Acid Make-up Tank Area
Access Hatch to Chem Waste Tank
Pressure Level Control Valve Rooms
Letdown Control Valve Rooms
Radioactive Pipeway
Valve Rooms
Letdown Control Valve Rooms
Pressure Level Control Valve Rooms
Caged Ladder Typ
Upper Part of Misc Waste Evap Cond Monitor Tank Rm
Upper Part of Misc Waste Evap Cond Monitor Tank Rm
Post Accident Sampling Room
Upper Part of Radwaste Primary Tank Room
Upper Part of Radwaste Primary Tank Room
Upper Part of Radwaste Primary Tank Room
Upper Part of Radwaste Primary Tank Room

FIGURE 7. Auxiliary Building Floor Plan (24.0-ft elevation) (showing measurement location Number 3).

BEST COPY AVAILABLE

21

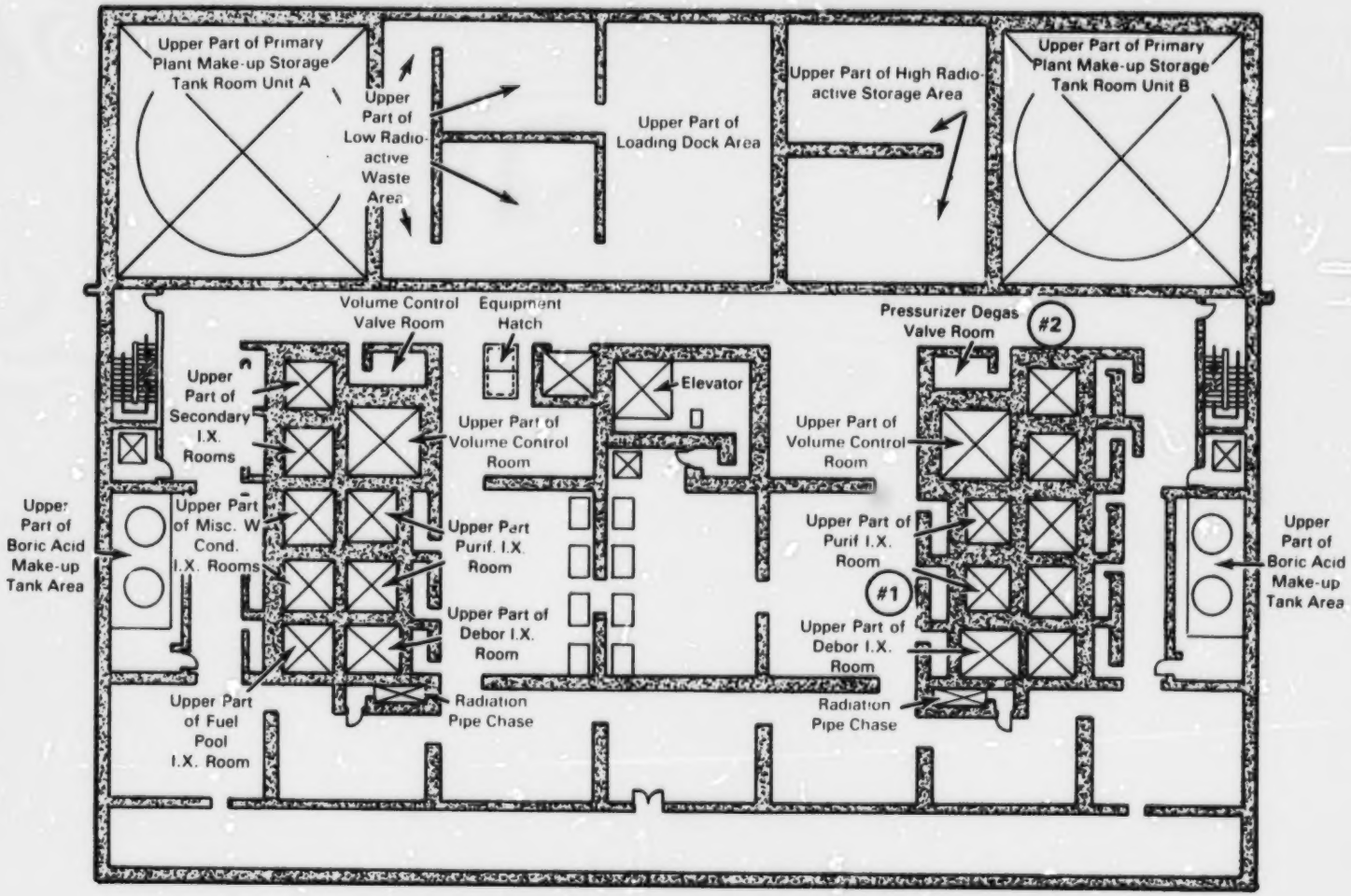

Upper Part of Primary Plant Make-up Storage Tank Room Unit A
Upper Part of Low Radio-active Waste Area
Upper Part of Loading Dock Area
Upper Part of High Radio-active Storage Area
Upper Part of Primary Plant Make-up Storage Tank Room Unit B
Volume Control Valve Room
Equipment Hatch
Pressurizer Degas Valve Room
#2
Upper Part of Secondary I.X. Rooms
Elevator
Upper Part of Volume Control Room
Upper Part of Volume Control Room
Upper Part of Misc. W Cond. I.X. Rooms
Upper Part Purif. I.X. Room
Upper Part of Purif I.X. Room
Upper Part of Boric Acid Make-up Tank Area
#1
Upper Part of Boric Acid Make-up Tank Area
Upper Part of Debor I.X. Room
Upper Part of Debor I.X. Room
Upper Part of Fuel Pool I.X. Room
Radiation Pipe Chase
Radiation Pipe Chase

FIGURE 8. Auxiliary Building Floor Plan (50.0-ft elevation) (showing measurement locations Number 1 and 2).

BEST COPY AVAILABLE

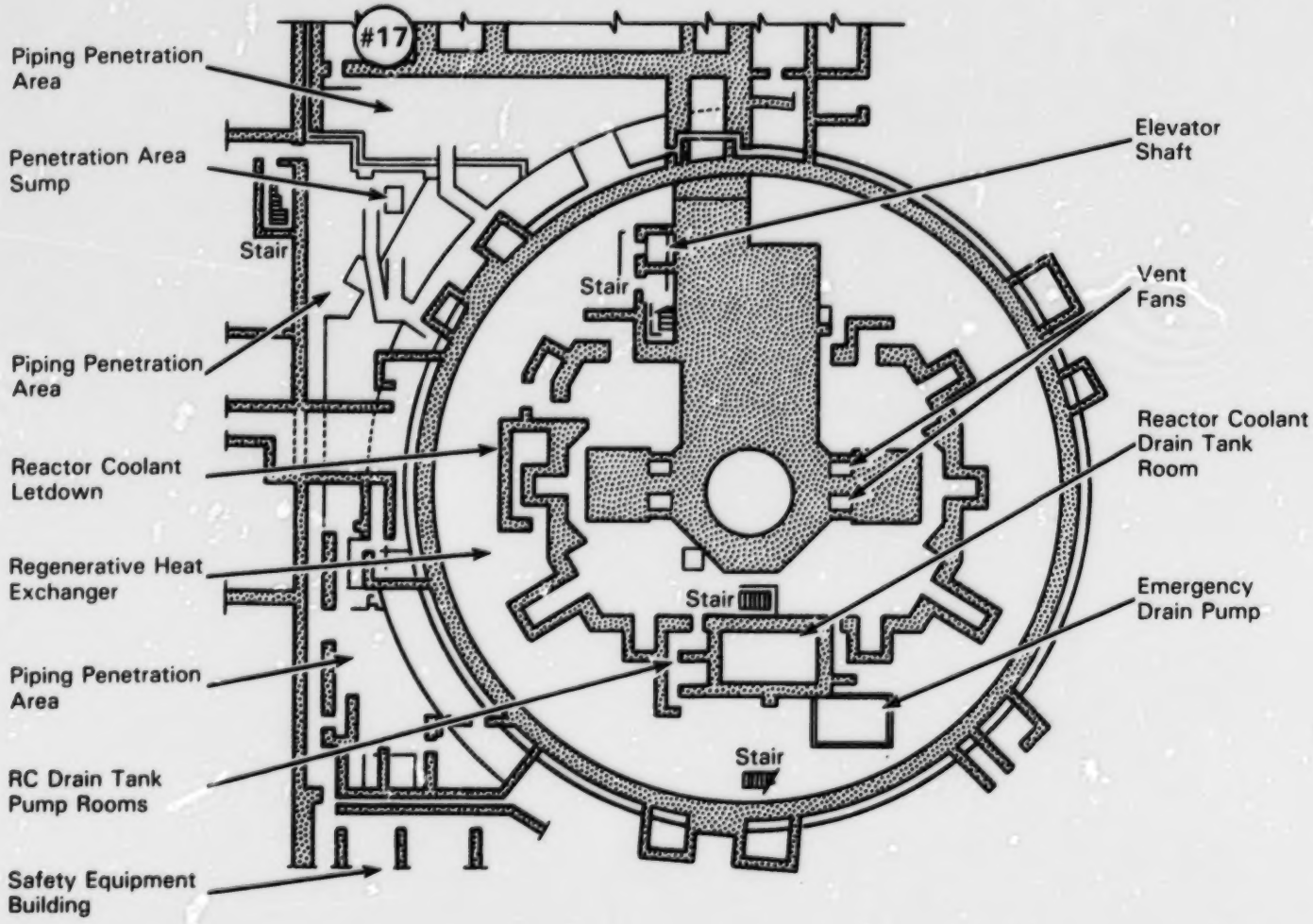


FIGURE 9. Containment Structure and Penetration Building Floor Plan (9.0-ft & 15.0-ft elevations) (showing measurement location Number 17).

BEST COPY AVAILABLE

23

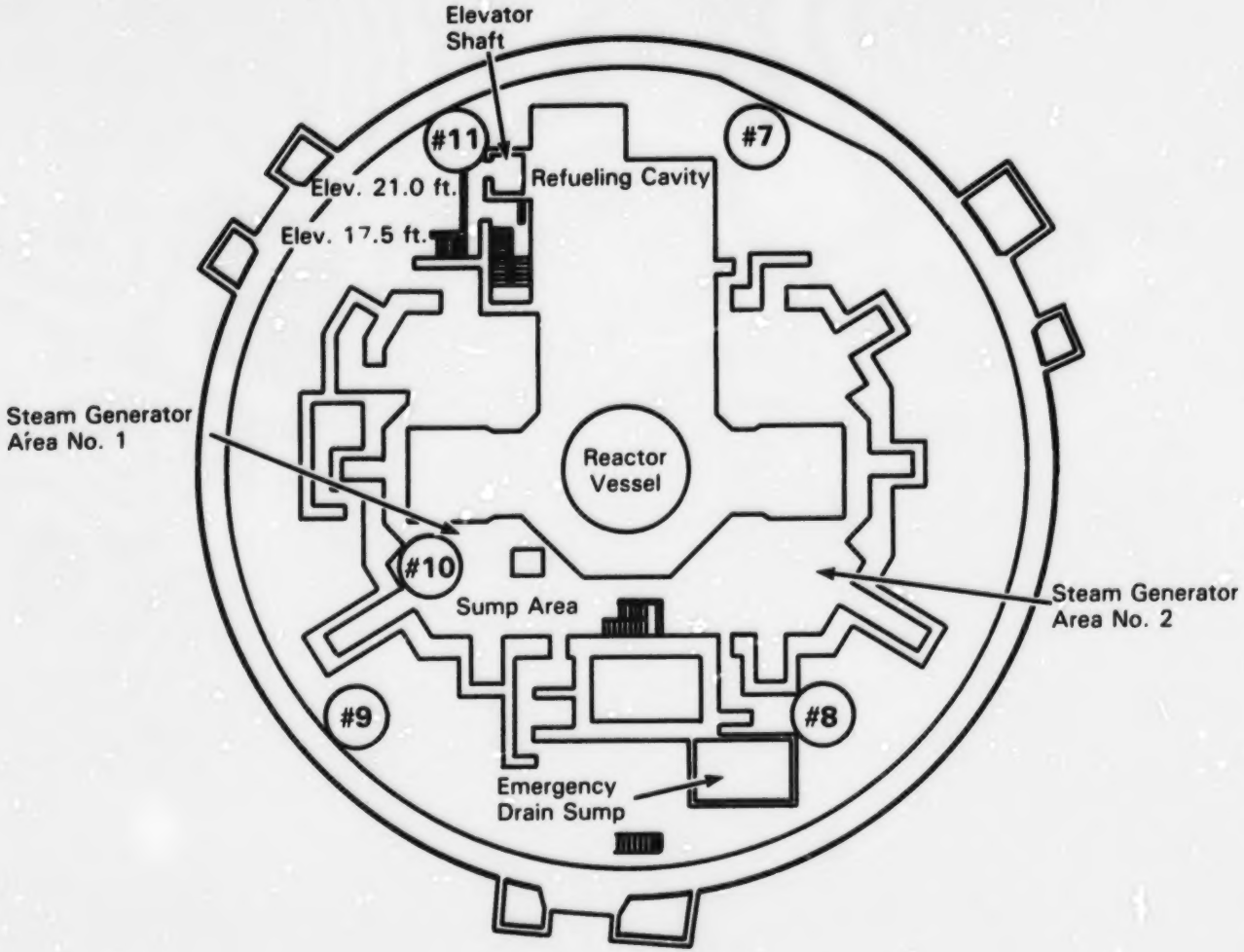


FIGURE 10. Containment Structure Floor Plan (17.5-ft elevation) (showing measurement locations Number 7, 8, 9, 10, and 11).

BEST COPY AVAILABLE

24

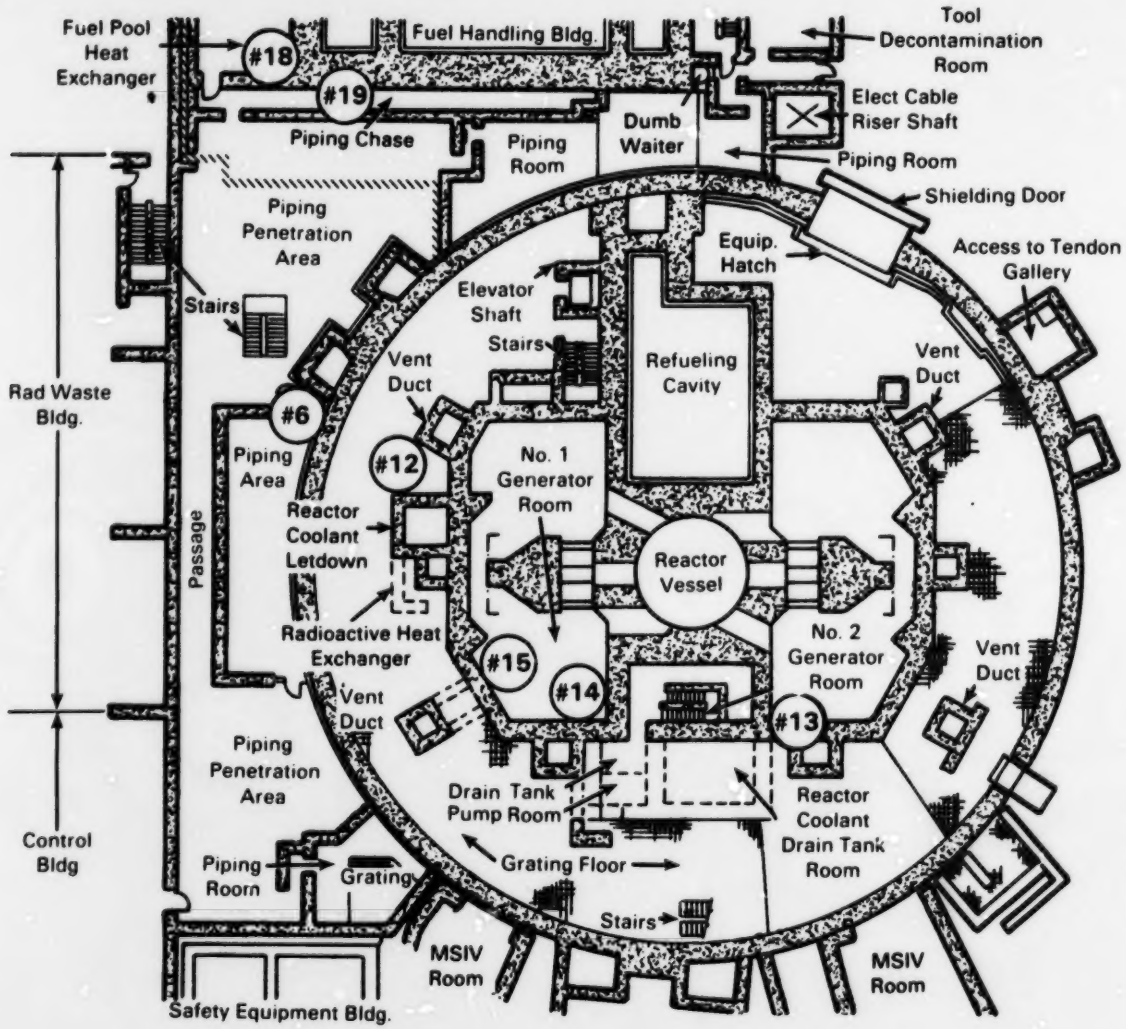


FIGURE 11. Containment Structure and Penetration Building Floor Plan (30.0-ft elevation) (showing measurement locations Number 6, 12, 13, 14, 15, 18, and 19).

BEST COPY AVAILABLE

25

BLANK PAGE

ANALYSIS AND DISCUSSION

The methodology and procedures presented in this report were designed to facilitate a quantitative evaluation of the impact of fuel cladding failures on occupational radiation exposures during a PWR maintenance and refueling outage. Data collection included gamma photon spectroscopy, radiation exposure rate measurements, airborne radionuclide concentrations, and beta radiation depth-dose measurements. Measurements were taken during a maintenance and refueling outage at a PWR following operation with fuel having excessive cladding failure.

Analysis of the data is intended to provide information on radiation exposure rates in specific areas of the plant as well as three functional catagories of plant systems and components. Measurements were made in nineteen separate locations and categorized according to the chemical and volume control system, the primary coolant system and components, and the refueling activities and spent fuel storage. In addition, individual radionuclide contributions to the exposure rates are evaluated in two ways. First, the radionuclide contributions to exposure rates are evaluated according to the individual radionuclides. The unique decay rate of each radionuclide results in a time-dependent contribution to exposure rate following reactor shutdown. Second, the radionuclide contributions to exposure rates are evaluated according to the contribution of activation versus fission products to determine their relative importance as a funtion of time.

Evaluation of the gamma photon spectral measuremerts included identification of radionuclides and analyses of the photon peaks to determine their individual contributions to exposure rates. Analyses of the gamma spectra began with the identification of all significant photon peaks using a Gaussian fit model. Many other less significant peaks were identified by a manual method. The radionuclide contributing to each peak was then identified by its characteristic energy and half-life. This procedure constituted the qualitative analyses. Analyses of the gamma spectral measurement data entered the quantitative stage with the application of a computer program to calculate the radiation exposure rate attributable to each radionuclide peak.

The computer code is essentially that described as KEVCAL.BAS in NUREG/CR-4485 Vol. 1 with several enhancements. To review, the program first determines the observed activity of each radionuclide using the total number of counts in the associated peak and the counting time. Next, the program accounts for the relative efficiency of the intrinsic gemanium detector as a function of photon energy. A new enhancement is that the program corrects for the relative efficiency of the portable ionization chamber, which tends to underestimate the exposure rate of gamma photons below 40 keV. This enhancement would not impact the results documented in NUREG/CR-4485 Vol. 1 because there were no significant peaks below 40 keV recorded. With this information, the program calculates the radiation exposure rate for each radionuclide based on an energy-dependent gamma dose conversion factor.

A second enhancement in the code permits comparison of the calculated total radiation exposure rate with the value input from the portable ioniza-

tion chamber measurement corrected for relative efficiency based on the spectrum of photon energies. This comparison allows the calculated total radiation exposure rate to be normalized to the corrected portable ionization chamber measurement. The fractional difference between the two measurements can be attributed to the physical characteristics of the measurement, including the position of the primary and secondary sources with respect to the detector. As a result, the fractional difference is listed on the computer analysis as the geometric correction factor. It is important to note that similar measurements tend to have essentially equal geometric correction factors.

With an estimate of the radiation exposure rate for each radionuclide peak calculated, the radiation exposure rate attributable to activation products, fission products. and other means of gamma and x-ray photon production can be determined. Unlike the comparison of comparable measurements made in two mirror-image units presented in NUREG/CR-4485 Vol. 1, measurements for this study were not collected with the intention of comparing data on the fuel cladding failure case with data on a baseline case. However, measurements were repeated in locations in the radwaste and primary penetration buildings in order to evaluate the decay of specific radionuclides over the course of the outage. New measurements in other locations were intended to indicate the extent of fission product contribution to radiation exposure rates at the time of the measurement.

Before the results of this study are presented, it is important to review the applicability of these data to other nuclear power plants. There are several features of the plant used in this study that are noteworthy in relating these findings to what might be expected at other plants. Because the measurements were taken at a PWR, it is obviously difficult to apply these data to a boiling-water reactor, which has a significantly different plant system configuration and layout. When relating these results to another PWR, it is necessary to consider that the measurements were taken at a relatively new plant with a 3640-megawatt-thermal (1100 MWe) power rating. Older and smaller plants would have different power ratings, construction materials, and reactor cleanup and purification systems that would directly affect the buildup of activation products and radiation exposure rates.

Another feature affecting the applicability of these data to another PWR is the previously described modification to the pressurizer steam space sample line. As noted, the modification provided continuous venting of the pressurizer's steam space to enhance removal of fission product gases from the reactor coolant system. This modification was necessary because of the high coolant activity concentrations present from the degraded fuel. As documented in NUREG/CR-4485 Vol. 1, the impact of this modification was to reduce the equilibrium coolant activity levels by an estimated 30 to 40 percent. However, it also resulted in the high specific activity condensate (1 to 2 mCi/cc) having been carried through piping from the steam space vent to areas of routine worker access. One such area is the volume control tank valve room where measurements were made during routine operation and repeated for this task. When considering how the data from this study could be applied to another PWR that does not have the pressurizer steam space vent (off-gas) line, it is necessary to consider what plant locations, systems, and components would have had greater radiation exposure rates entering into the maintenance and refueling

outage as a result of the higher coolant activity and radiation levels from the steam space effluent.

RESULTS

In the text that follows, the results of the measurements obtained in this study are presented in relation to plant systems and components. A summary of the results is presented in Table 3. The repeat measurements on the plant chemical and volume control system are presented first. Systems on this pathway include purification ion-exchangers and charging pumps. These results are presented following the sequence of coolant flow from the containment structure through the system components, and finally as it re-enters containment. Measurements taken inside are the reactor containment building on components of the primary coolant system presented second. Systems and components inside containment include the pressurizer, steam generators, letdown valves, and shutdown cooling lines. As a third and final presentation, measurements are given relative to refueling activities and spent fuel storage. These include two measurements taken inside containment on the refueling cavity drain line and three measurements taken in the spent fuel pool pump, heat-exchanger, and piping area of the penetration building. Systems and components in this area are designed to purify and recirculate cooling water through the spent fuel pool. Because of contamination with the small highly-radioactive fuel particles, the fuel handling building was not accessible. Radiation exposure rates on the fuel handling deck inside containment were not considered significant enough to warrant gamma spectroscopy measurements. For the purpose of presenting the results, each measurement is identified by location and by the PWR outage measurement number (POM#).

Before presenting the gamma photon spectral measurement data and associated radiation exposure estimates, it is noteworthy to review the TLD exposures and air sampling data. The TLD data indicate that there was no significant contribution from beta radiation to occupational radiation exposures at the measured locations. This was evident because the readouts on the chip with a thin mylar covering and the chip with a thicker 125-mil (0.125 in.) aluminum covering are equivalent. In addition to determining the lack of a beta radiation contribution to exposure rate, the TLD exposures were used as an indicator of the gamma photon field in support of the ionization chamber measurements. Analyses of the filter papers and silver-zeolite cartridges from collected air samples indicated that airborne radionuclides were not present at any of the measurement locations. This result is consistent with the routine air sample data collected by the plant licensee during the outage.

An important aspect in evaluating the radionuclide contributions to exposure rates is to determine how they change as a function of time. An analysis of the changing source term can be made through a comparison of measurements made during routine operations shortly before the outage and those repeated during the outage. Comparison of these gamma spectral measurements indicate that the radionuclide composition of the measured source terms changes significantly from the beginning to the end of the outage as a result of differences in the decay rates of individual radionuclides. Because the outage measurements

TABLE 3. Summary of Onsite Measurement Results

PWR Outage Measurement Number	Measurement Area Description	Exposure Rate at Detector* (mR/h)	Percentage (%) of Total Exposure Rate		
			Activation Products	Fission Products	Other Means of Photon Production
	CHEMICAL AND VOLUME CONTROL SYSTEM		25.0 ± 12.4	43.8 ± 16.0	31.3 ± 5.6
6	Inside Pipe Penetration Room	1.8	10.3	65.5	24.3
1	Purifying Ion-exchanger Valve Room/Inlet	0.3	41.4	24.8	33.8
4	Bypass Ion-exchanger Valve Room/Outlet	2.1	17.4	58.2	24.4
2	Pressurizer Off-gas Line	0.2	38.0	29.4	32.6
3	Pressurizer Level Control Valve Room	1.6	25.5	38.6	36.0
5	Charging Pump Room	0.1	17.2	46.1	36.7
	PRIMARY COOLANT SYSTEM AND COMPONENTS		44.9 ± 12.3	43.5 ± 18.4	11.7 ± 11.6
16	Pressurizer Surge Line	8.1	46.2	42.5	11.4
13	Hot Leg of Steam Generator #2	5.0	56.8	39.6	3.6
8	Valve Near Steam Generator #2	6.0	50.3	46.3	3.4
10	Letdown Valve Near Steam Generator #1	5.1	60.4	12.6	27.1
9	Shutdown Cooling Line - 17'6"-Level	6.5	39.0	57.3	3.8
12	Shutdown Cooling Line - 30'-Level	4.1	22.3	76.4	1.3
15	Shutdown Cooling Line Upstream of POM14 Valve	4.3	40.7	48.1	11.2
14	Shutdown Cooling Isolation Valve	6.2	35.7	32.5	31.8
	REFUELING ACTIVITIES AND SPENT FUEL STORAGE		52.2 ± 15.6	32.0 ± 19.6	15.8 ± 5.9
11	Cavity Drain Line Near Elevation Access Point	4.1	56.6	23.6	19.8
7	Cavity Drain Line At Termination	4.1	72.9	3.6	23.5
17	Spent Fuel Pool Pump Room	3.0	48.0	38.8	13.1
18	Fuel Pool Heat Exchanger	2.6	29.7	55.9	14.4
19	Fuel Pool Heat Exchanger - Main Line	7.0	53.8	37.9	8.4

* Corrected for the efficiency of the portable ionization chamber as a function of photon energy.

BEST COPY AVAILABLE

30

were taken about eighty days following shutdown, the presence of a short-lived radionuclide with a decay half-life of less than approximately ten days would not be expected unless the radionuclide were a product of the decay of a longer-lived parent.

Analysis of the gamma spectral data indicate the presence of 511 keV photons and apparent low-energy photopeaks. Two known sources exist for the 511 keV photons. First, two of the radionuclides identified as being present are positron emitters (Co-58 and Zn-65); thus 511 keV photons are produced when the positrons are annihilated. Second, a large number of the photons measured were greater than 1022 keV which means that 511 keV photons could also be produced by the annihilation of positrons produced by pair production.

An explanation for the apparent low-energy photopeaks is not as definitive as that for the 511 keV photons. There are three potential explanations for the recurring photopeak spanning from approximately 26 to 33 keV in each of the spectra. The magnitude of the peak does vary significantly between different locations. First, it is possible that the peak is not a peak at all, but an artifact of the discriminator level. In other words, the peak is not the result of photons but rather electronics. Figures B.2 and B.6 appear to indicate that the low-energy photopeak is not an artifact of the discriminator level. Second, the apparent low-energy x rays may be produced by K-shell emissions from atoms comprising the metal piping and components. These metals include silver (Ag), cadmium (Cd), indium (In), tin (Sn), antimony (Sb), and tellurium (Te). These K-shell x rays have an average energy of approximately 29.4 keV. Third, the apparent low-energy x rays may be produced by exitation of materials within the detector. Many of the metals identified in the piping and components are also in the detector. Because the exact source of the recurring low-energy peak is not known and there is not enough evidence to disregard the data, the peak is identified on each spectrum and the associated analysis as merely "unknown". In terms of analysis, the exposure rate contributed by these peaks is included in the category labelled "Other Means of Gamma Photon and X Ray Production" and does not impact the activation versus fission product ratio nor the conclusions derived from evaluation this ratio. Within the discussion of each measurement location, reference may be made to relative magnitude of the apparent low-energy photopeaks. These references are not intended to lend more credence to explanations of the photopeaks involving low-energy x rays than the explanation of the photopeaks as an artifact of the disciminator level.

CHEMICAL AND VOLUME CONTROL SYSTEM

The following six measurements were taken on the chemical and volume control system in the radwaste and pipe penetration building. These measurements were a repeat of measurements taken during routine operations. The results are presented following the sequence of coolant flow from the containment structure, through the system components, and finally as it re-enters containment.

Inside Primary Pipe Penetration Room (POM06)

Analyses of the gamma spectrum and exposure rate measurements suggest that, inside the primary pipe penetration room, fission products contributed

more to occupational exposures than did activation products. Activation products, fission products, and other means of gamma photon or x-ray production were responsible for approximately 10, 65, and 24 percent of the total exposure rate, respectively. The dominant fission product radionuclides were Cs-137 (cesium) and Cs-134. The dominant activation product radionuclides were Co-60 (cobalt) and Co-58. The apparent low-energy x rays contributed significantly to the exposure rate (24%). The radiation exposure rate as measured with a portable ionization chamber had decreased from 3.5 mR/h during routine operations to 1.7 mR/h at the time of the outage measurement. Important short-lived radionuclides that had apparently decayed below detectable levels included Kr-88 (krypton), Rb-88 (rubidium), Xe-135 (xenon), and Na-24 (sodium) as well as isotopes of iodine (I).

Purifying Ion-exchanger Valve Room/Inlet (POM01)

Evaluation of the gamma spectral and exposure rate measurements indicate that activation products contributed more to occupational exposures than did fission products in the purifying ion-exchanger valve room. Activation products, fission products, and other means of production were responsible for approximately 41, 25, and 34 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-60, Mn-54 (manganese), and Co-58. The dominant fission product radionuclides were Cs-137 and Cs-134. The apparent low-energy x rays contributed significantly to the exposure rate (32%). The radiation exposure rate as measured with a portable ionization chamber had decreased from 0.5 mR/h during routine operations to 0.3 mR/h at the time of the outage measurement. Important short-lived radionuclides that had apparently decayed below detectable levels included Kr-88, Rb-88, and Xe-135 as well as isotopes of iodine.

Bypass Ion-exchanger Valve Room/Outlet (POM04)

Based on gamma spectral and exposure rate measurements, fission products contributed more to occupational exposure than did activation products at this location. Activation products, fission products, and other means of gamma photon and x-ray production were responsible for approximately 17, 58, and 24 percent of the total exposure rate, respectively. The dominant fission product radionuclides were Cs-137, Cs-134, and Nb-95 (niobium). The dominant activation product radionuclides were Co-60, Co-58, and Mn-54. The apparent low-energy x rays contributed significantly (23%) to the exposure rate. The radiation exposure rate as measured with a portable ionization chamber had decreased from 26 mR/h during routine operations to 2.0 mR/h at the time of the outage measurement. Important short-lived radionuclides that had apparently decayed below detectable levels included Kr-88, Rb-88, Cs-138, Kr-87, Na-24, Xe-138, and I-137.

Off-Gas Line from Pressurizer Located in the Volume Control Tank Room - Plant Modification (POM02)

Analyses of the gamma spectral and exposure rate measurements suggest that activation products contributed more to occupational exposure than did fission products at this location. Activation products, fission products, and

other means of production were responsible for approximately 38, 29, and 33 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-60, Co-58, and Mn-54. The dominant fission product radionuclides were Cs-137, Cs-134, Nb-95, and Zr-95 (zirconium). The apparent low-energy x rays contributed significantly (31%) to the exposure rate. At this location, there was a significant peak associated with the 511 keV annihilation energy. The radiation exposure rate as measured with a portable ionization chamber had decreased from 29 mR/h during routine operations to 0.2 mR/h at the time of the outage measurement. Important short-lived radionuclides that had apparently decayed below detectable levels included Kr-88, Rb-88, Na-24 and Kr-87. It is worth noting that although Zr-95 is both an activation and a fission product, the production of Zr-95 by activation [Zr-94(n,γ)Zr-95] is negligible when compared to the fission product yield of Zr-95. Zirconium is a metal used in the cladding of the enriched uranium fuel.

Pressurizer Level Control Valve Room (POM03)

Evaluation of the gamma spectral and exposure rate measurements indicate that fission products contributed more to occupational exposure than did activation products at this location. Activation products, fission products, and other means of production were responsible for approximately 25, 39, and 36 percent of the total exposure rate, respectively. The dominant fission product radionuclides were Cs-137, Cs-134, Nb-95, and Zr-95. The dominant activation product radionuclides were Co-60, Co-58, Mn-54, and Zn-65 (zinc), as well as Sn-113. The pressurizer level control valve room is the only area to have Zn-65 as a dominant activation product. The apparent low-energy x rays contributed significantly (34%) to the exposure rate. At this location, there was a significant peak associated with the 511 keV annihilation energy. These measurements were a repeat of measurements taken during routine operations on the boric acid make-up system. The reason for the labelling change is that the boric acid make-up system is a once-through process and does not recirculate coolant. Consequently, the boric acid make-up system cannot be radioactively contaminated. The source of contamination at this location is the pressurizer level control valve. Despite the labelling change, comparison of the outage measurement with that taken prior to the outage facilitates an evaluation of the changing source term and contributing radionuclides. The radiation exposure rate as measured with a portable ionization chamber had decreased from 11 mR/h during routine operations to 1.5 mR/h at the time of the outage measurement. Important short-lived radionuclides that had apparently decayed below detectable levels included Kr-88, Rb-88, Cs-138, Xe-138, and Na-24.

Charging Pump Room (POM05)

Based on the data collected using the intrinsic Ge detector for gamma spectroscopy and the ionization chamber for exposure rate measurements, in the charging pump room, fission products contributed more to occupational exposure than did activation products. Activation products, fission products, and other means of production were responsible for approximately 17, 46, and 37 percent of the total exposure rate, respectively. The dominant fission product radionuclides were Cs-137, Cs-134, Nb-95, and Zr-95. The dominant activation product radionuclides were Co-60, Co-58, and Mn-54. The apparent low-energy

x-rays contributed significantly (37%) to the exposure rate. The radiation exposure rate as measured with a portable ionization chamber had decreased from 3.1 mR/h during routine operations to 0.1 mR/h at the time of the outage measurement. Important short-lived radionuclides that had apparently decayed below detectable levels included Kr-87, Kr-88, Rb-88, and Mn-56 as well as isotopes of iodine and xenon.

PRIMARY COOLANT SYSTEM AND COMPONENTS

The following eight measurements were taken inside the containment building on the primary coolant system and its associated components. The systems and components include the pressurizer, steam generators, letdown valves, and shutdown cooling lines. When measurements were taken during routine operations at the plant, the inside of the containment building was not accessible because of high radiation exposure rates and airborne activity. Consequently, unlike the chemical and volume control measurements, there are no previous measurements for comparison.

Pressurizer Surge Line (POM16)

Analyses of the gamma spectral and exposure rate measurements suggest that, near the pressurizer surge line, activation products contributed more to occupational exposure than did fission products. Activation products, fission products, and other means of gamma photon and x-ray production were attributable for approximately 46, 42, and 11 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-60, Co-58, and Mn-54. The dominant fission product radionuclides were Cs-137, Cs-134, Nb-95, and Zr-95. The apparent low-energy x rays contributed significantly (10%) to the exposure rate. At this location, there was a significant peak associated with the 511 keV annihilation energy. The radiation exposure rate as measured with the portable ionization chamber was 8.0 mR/h at the time of the outage measurement.

Hot Leg of Steam Generator #2 (POM13)

Evaluation of the gamma spectral and exposure rate measurements indicated that, near the hot leg of steam generator #2, activation products contributed more to occupational exposure than did fission products. Activation products, fission products, and other means of gamma photon and x-ray production were responsible for approximately 57, 40, and 4 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-60, Co-58, and Mn-54. The dominant fission product radionuclides were Cs-137, Cs-134, and Nb-95. The apparent low-energy x rays contributed less significantly (3%) to the exposure rate near the hot leg of steam generator #2 than at other locations. At this location, there was a significant peak associated with the 511 keV annihilation energy. The radiation exposure rate as measured with the portable ionization chamber was 5.0 mR/h at the time of the outage measurement.

Valve Near Steam Generator #2 (POM08)

Based on the data collected using the intrinsic germanium detector for gamma spectroscopy and the ionization chamber for exposure rate measurements,

at this location, activation products contributed more to occupational exposure than did fission products. Activation products, fission products, and other means of production were responsible for approximately 50, 46, and 3 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-60, Mn-54, and Co-58. The dominant fission product radionuclides were Cs-137, Cs-134, and Nb-95. Similar to the other measurement near steam generator #2, the apparent low-energy x rays contributed less significantly (3%) to the exposure rate here than at other locations. The radiation exposure rate as measured with the portable ionization chamber was 6.0 mR/h at the time of the outage measurement.

Letdown Valve Near Steam Generator #1 (POM10)

Analyses of the gamma spectral and exposure rate measurements suggest that, at this location, activation products contributed more to occupational exposure than did fission products. Activation products, fission products, and other means of production were responsible for approximately 60, 13, and 27 percent of the total exposure rate, respectively. The dominant activation product radionuclides included Co-60, Co-58, and Mn-54, as well as Fe-59 (iron). The dominant fission product radionuclides were Nb-95, Zr-95, Cs-137, and Cs-134. The measurements at this location are unusual in that the Zr-95 and its decay product, Nb-95, contributed more to the exposure rate than did the isotopes of cesium. The apparent low-energy x rays contributed significantly (27%) to the exposure rate. At this location, there was a significant peak associated with the 511 keV annihilation energy. The radiation exposure rate as measured with the portable ionization chamber was 5.0 mR/h at the time of the outage measurement.

Shutdown Cooling Line - 17' 6" Level (POM09)

Evaluation of the gamma spectral and exposure rate measurements indicate that fission products contributed more to occupational exposure than did activation products at this location. Activation products, fission products, and other means of production were responsible for approximately 39, 57, and 4 percent of the total exposure rate, respectively. The dominant fission product radionuclides were Cs-134, Cs-137, Nb-95, and Zr-95. The dominant activation product radionuclides were Co-60, Co-58, and Mn-54, as well as Fe-59. The apparent low-energy x rays contributed less significantly (2%) to the exposure rate here than at other locations. At this location, there was a significant peak associated with the 511 keV annihilation energy. The radiation exposure rate as measured with the portable ionization chamber was 6.5 mR/h at the time of the outage measurement.

Shutdown Cooling Line - 30' Level (POM12)

Based on the data collected using the intrinsic germanium detector for gamma spectroscopy and the ionization chamber for exposure rate measurements, fission products contributed more to occupational exposure than did activation products near the shutdown cooling line on the 30-foot level inside containment. Activation products, fission products, and other means of production were responsible for approximately 22, 76, and 1 percent of the total exposure rate, respectively. The dominant fission product radionuclides were Cs-137

and Cs-134. The Cs-137 peak at 661.6 keV contributed an estimated 36 percent of the measurable exposure rate. The dominant activation product radionuclides were Co-60, Co-58, and Mn-54. The apparent low-energy x rays did not contribute significantly (<1%) to the exposure rate. The radiation exposure rate as measured with the portable ionization chamber was 4.1 mR/h at the time of the outage measurement.

Shutdown Cooling Line Upstream of POM14 Valve (POM15)

The next two measurements were designed to provide a comparison between a suspected radioactive crud trap (POM14) and a section of piping feeding into that location (POM15). Analyses of the gamma spectral and exposure rate measurements suggest that activation products contributed more to occupational exposure than did fission products in the piping upstream of the radioactive crud trap. Activation products, fission products, and other means of production were responsible for approximately 48, 41, and 11 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-60, Co-58, and Mn-54. The dominant fission product radionuclides were Cs-137, Cs-134, and Nb-95. The apparent low-energy x rays contributed significantly (10%) to the exposure rate. At this location, there was a significant peak associated with the 511 keV annihilation energy. The radiation exposure rate as measured with the portable ionization chamber was 4.2 mR/h at the time of the outage measurement.

Shutdown Cooling Isolation Valve (POM14)

This shutdown cooling isolation valve was downstream of the section of piping used as measurement location POM15. Evaluation of the gamma spectral and exposure rate measurements indicate that activation products contributed more to occupational exposure than did fission products at this suspected radioactive crud trap. Activation products, fission products, and other means of production were responsible for approximately 36, 32, and 32 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-60, Co-58, and Mn-54. The dominant fission product radionuclides were Cs-137 and Cs-134. The apparent low-energy x rays contributed significantly (31%) to the exposure rate. Data from the measurements suggest a similar radionuclide composition both at and upstream of the valve. However, there was more activity from Co-58, Fe-59, and Cs-137 upstream of the crud trap. This suggests, perhaps, that these radionuclides pass through the crud trap in the valve more effectively than other radionuclides in the source term. The major difference between the two measurement points appears to be the relative contribution by the apparent low-energy x rays at the valve (31%) versus upstream in the pipe (10%). The radiation exposure rate as measured with the portable ionization chamber was 6.0 mR/h at the time of the outage measurement.

REFUELING ACTIVITIES AND SPENT FUEL STORAGE

The following five measurements were intended to provide data on exposure rates involving refueling activities and spent fuel storage. These include two measurements taken inside containment on the main cavity drain line and three measurements taken in the spent fuel pool pump, heat-exchanger, and

piping area of the penetration building. Systems and components in this area are designed to purify and recirculate cooling water through the spent fuel pool. The ideal situation would have been to take measurements in the fuel handling building. Unfortunately, as described previously, the fuel handling building was contaminated with the small highly-radioactive fuel particles and, therefore, was not accessible. Although accessible, radiation exposure rates on the fuel handling deck inside containment were not considered significant enough to warrant gamma spectroscopy measurements.

Refueling Cavity Drain Line Near Elevation Access Point (POM11)

Based on the data collected using the intrinsic germanium detector for gamma spectroscopy and the ionization chamber for exposure rate measurements, at this location, activation products contributed more to occupational exposure than did fission products. Activation products, fission products, and other means of gamma photon and x-ray production were responsible for approximately 57, 24, and 20 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-60, Co-58 and Mn-54. The dominant fission product radionuclides were Cs-137, Cs-134, Nb-95, and Zr-95. The apparent low-energy x rays contributed significantly (18%) to the exposure rate. At this location, there was a significant peak associated with the 511 keV annihilation energy. The radiation exposure rate as measured with the portable ionization chamber was 4.0 mR/h at the time of the outage measurement.

Cavity Drain Line At Termination (POM07)

Analyses of the gamma spectral and exposure rate measurements suggest that activation products contributed more to occupational exposure than fission products at this location. Activation products, fission products, and other means of gamma photon and x-ray production were attributable for approximately 73, 4, and 23 percent of the total exposure rate, respectively. The dominant activation product radionuclides included Co-60, Co-58, and Mn-54, as well as Cr-51 (chromium) and Fe-59. The cavity drain line at termination is the only location where Cr-51 is a notworthy radionuclide. The dominant fission product radionuclide was Nb-95. It is worth noting that this location did not have a significant contribution to exposure rate from Cs-137 or Cs-134. The apparent low-energy x rays contributed significantly (22%) to the exposure rate. At this location, there was a significant peak associated with the 511 keV annihilation energy. The radiation exposure rate as measured with the portable ionization chamber was 4.0 mR/h at the time of the outage measurement. Due to the high radiation exposure rate near the main cavity drain line, the measurement was made approximately 30 feet from the primary source.

Spent Fuel Pool Pump Room (POM17)

Evaluation of the gamma spectral and exposure rate measurements indicate that, in the spent fuel pool pump room, activation products contributed more to occupational exposure than did fission products. Activation products, fission products, and other means of production were responsible for approximately 48, 39, and 13 percent of the total exposure rate, respectively. The dominant activation product radionuclides were Co-58, Co-60, and Mn-54, as

well as Fe-59. The Co-58 contributed an estimated 24 percent to the total exposure rate. The dominant fission product radionuclides were Cs-134, Cs-137, Nb-95, and Zr-95. The apparent low-energy x rays contributed significantly (8%) to the exposure rate. At this location, there was a very prominant peak associated with the 511 keV annihilation energy contributing significantly (5%) to the exposure rate. One explanation is that the principal radionuclide, Co-58, is a positron emitter. The radiation exposure rate as measured with the portable ionization chamber was 3.0 mR/h at the time of the outage measurement.

Fuel Pool Heat Exchanger (POM18)

Based on the data collected using the intrinsic Ge detector for gamma spectroscopy and the ionization chamber for exposure rate measurements, fission products contributed more to occupational exposure than did activation products near the fuel pool heat exchanger. Activation products, fission products, and other means of production were responsible for approximately 30, 56, and 14 percent of the total exposure rate, respectively. The dominant fission product radionuclides were Cs-137, Cs-134 and Nb-95. The dominant activation product radionuclides were Co-58 and Co-60, as well as Mn-54. The Co-58 peak at 810.8 keV contributed an estimated 19 percent of the measurable exposure rate. The apparent low-energy x rays contributed significantly (11%) to the exposure rate. The radiation exposure rate as measured with the portable ionization chamber was 2.6 mR/h at the time of the outage measurement.

Fuel Pool Heat Exchanger - Main Line (POM19)

Analyses of the gamma spectral and exposure rate measurements suggest that, at this location, activation products contributed more to occupational exposure than did fission products. Activation products, fission products, and other means of production were responsible for approximately 54, 38, and 8 percent of the total exposure rate, respectively. The dominant activation product radionuclides included Co-58, Co-60, Mn-54, and Fe-59, as well as Cr-51. The Co-58 peak at 810.8 keV contributed an estimated 24 percent of the measurable exposure rate. The dominant fission product radionuclides were Nb-95, Cs-134, Cs-137, and Zr-95. The contribution to exposure rate from Nb-95 and Zr-95 is particularly high in this location as compared to other areas of the plant. The apparent low-energy x rays contributed moderately (4%) to the exposure rate. At this location, there was a significant peak associated with the 511 keV annihilation energy. As previously described, this relatively high contribution (4%) to exposure rate from 511 keV photons could be related to the fact that Co-58 is a positron emitter. The radiation exposure rate as measured with the portable ionization chamber was 7.0 mR/h at the time of the outage measurement. All three measurements near the spent fuel pump room had a significant Co-58 contribution.

RADIONUCLIDE CONTRIBUTIONS

Analysis of the data has provided information on radiation exposure rates in specific areas according to three functional catagories of plant systems and components. The results presented in the previous section have emphasized

the specific locations rather than general trends. Listed below are results relating radionuclide contributions to specific plant systems.

There exists:

- a greater contribution to exposure rate from activation products in the refueling activities and spent fuel storage than in the chemical and volume control system;
- a greater contribution to exposure rate from activation products in the primary coolant system and compontents than in the chemical and volume control system;
- a greater contribution to exposure rate from Co-58 in areas associated with the spent fuel storage than elsewhere in the plant;
- a greater contribution to exposure rate from the 511 keV annihilation peak in the refueling activities and spent fuel storage than in the primary coolant system and components or in the chemical and volume control system (This appears to be related to the significant amount of Co-58, which is a positron emitter, in areas associated with the spent fuel storage);
- a greater contribution to exposure rate from Nb-95 and Zr-95 in the refueling activities and spent fuel storage than in the primary coolant system and components or in the chemical and volume control system;
- a greater contribution to exposure rate from the apparent low-energy x rays in the chemical and volume control system than in the primary coolant system and components or in the refueling activities and spent fuel storage;
- a lesser contribution to exposure rate from the apparent low-energy x rays near steam generator Number 2 and in the shutdown cooling lines than in other areas
- a greater contribution to exposure rate from Co-60 in the cavity drain lines and near steam generator Number 2 than in other areas of the plant
- a lesser contribution to exposure rate from Co-60 in the spent fuel pool heat exchangers than in other areas of the plant
- a greater contribution to exposure rate from Cs-137 in the shutdown cooling lines and in the pipe penetration room than in other areas of the plant
- a lesser contribution to exposure rate from Cs-134 and Cs-137 in the cavity drain lines and letdown valve than in other areas of the plant.

ESTIMATED EXPOSURE RATES AFTER REACTOR SHUTDOWN

A comparison of the gamma spectral measurements made during routine operations shortly before the outage with those repeated during the outage serve two purposes. First, as previously stated, the gamma spectral measurements indicate that the radionuclide composition of the measured source terms changes significantly from the beginning to the end of the outage. This change is the result of differences in the decay rates of individual radionuclides. Second, such comparisons would be useful in determining the extent to which the relative contribution to radiation exposure rates attributable to activation versus fission products changes from the beginning to the end of the outage. The remainder of this section presents the calculations used to estimate attributable exposure rates at different times after reactor shutdown.

Calculations have been made of the estimated exposure rates at different times after reactor shutdown relative to the measurements made inside the primary pipe penetration room during routine operations. A summary of these calculations is presented in Table 4. Detailed calculations at different time periods (i.e., t = 0, 2, 5, 10, 20, 40, and 80 days) after plant shutdown are included in Appendix D of this report. Table 4 lists the estimated exposure rates and percentage of total exposure rate attributable to activation products, fission products, and other means of x ray and gamma photon production. The contribution from the other means of production to the total exposure rate remained at low levels throughout this time period. The tables in Appendix D list data specific to each radionuclide present. Also included in Appendix D are data obtained in the pipe penetration room during the outage with the apparent low-energy photon (unknown) contribution eliminated so that direct measurement data could be compared to the calculated estimates. The exposure rates present during routine operations were measured several weeks prior to plant shutdown and were assumed to be the same as the exposure rates present at the time the plant was shutdown. The calculated exposure rates at the different times take into account only the radioactive decay of each radionuclide; no other parameters or assumptions have been included.

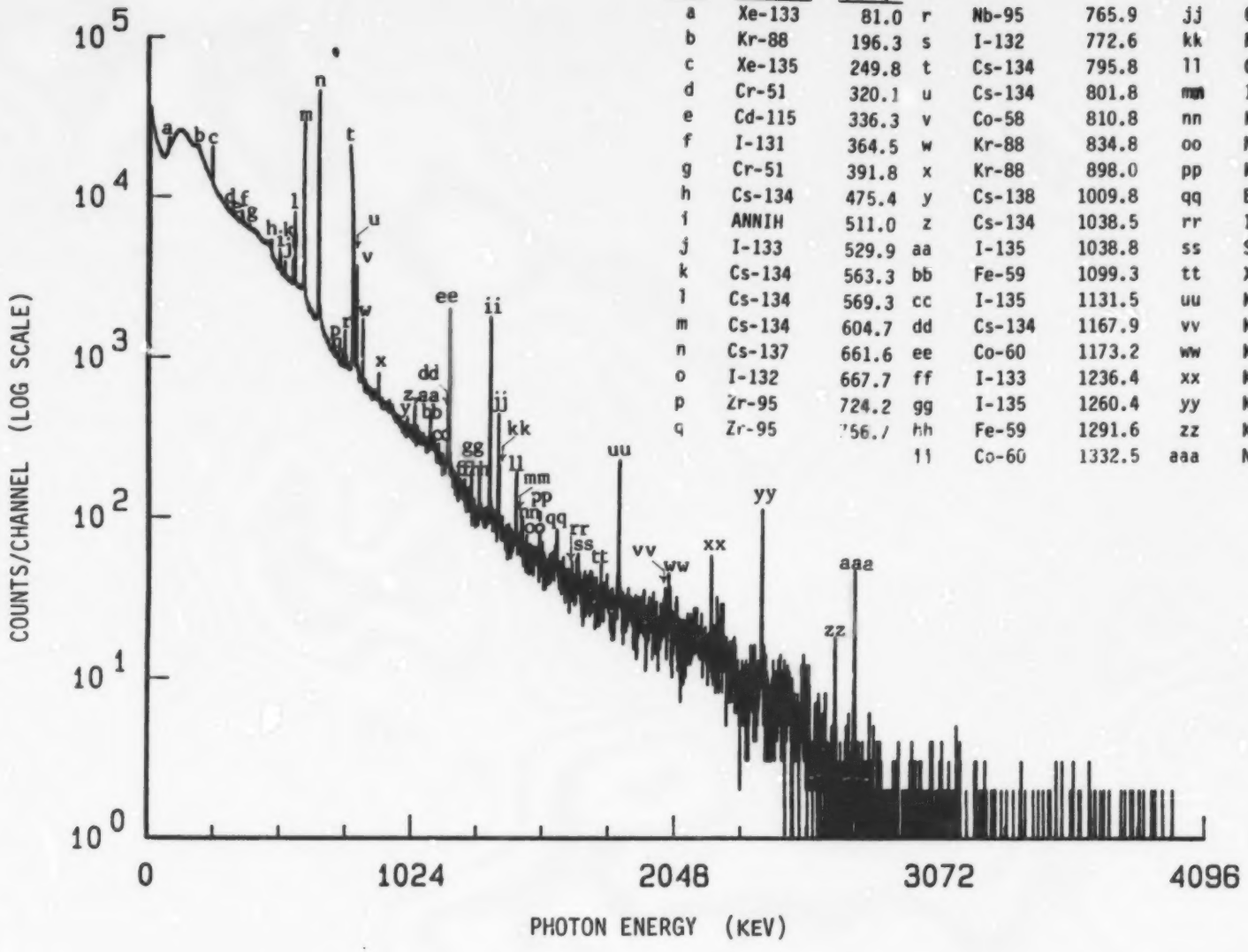
Figures 12 and 13 represent the gamma spectra obtained inside the primary pipe penetration room during routine operations and the gamma spectra obtained approximately eighty days after reactor shutdown, respectively. Each is plotted over 4096 channels calibrated to 1 keV per channel. Two different intrinsic germanium detectors were used for the two sets of measurements. The main difference between the two spectra is the absence of short-lived radionuclides in the outage measurement. The large number of apparent low-energy photons which were measured during the outage measurement were not detected during the routine operations. Two possible reasons for this difference are that these x rays were masked by the higher exposure rates present during routine operations and that these x rays were measured differently due to differences in the equipment used during the two studies.

TABLE 4. Estimated Exposure Rates for Individual Radiation Sources and Percentage of Total Exposure Rate for Selected Times After Plant Shutdown

Time (Days)	Exposure Rate (mR/h)			% of Total Dose Rate		
	Activation Products	Fission Products	Other Means of Gamma Photon and X Ray Production	Activation Products	Fission Products	Other Means of Gamma Photon and X Ray Production
0	0.45	3.02	3.3×10^{-2}	12.85	86.20	0.95
2	0.42	2.75	2.7×10^{-2}	13.00	86.10	0.85
5	0.41	2.74	2.7×10^{-2}	12.85	86.30	0.85
10	0.40	2.72	2.7×10^{-2}	12.72	86.4	0.85
20	0.38	2.70	2.7×10^{-2}	12.29	86.86	0.85
40	0.37	2.66	2.6×10^{-2}	12.00	87.10	0.85
80	0.38	2.60	2.5×10^{-2}	11.40	87.8	0.86

BEST COPY AVAILABLE

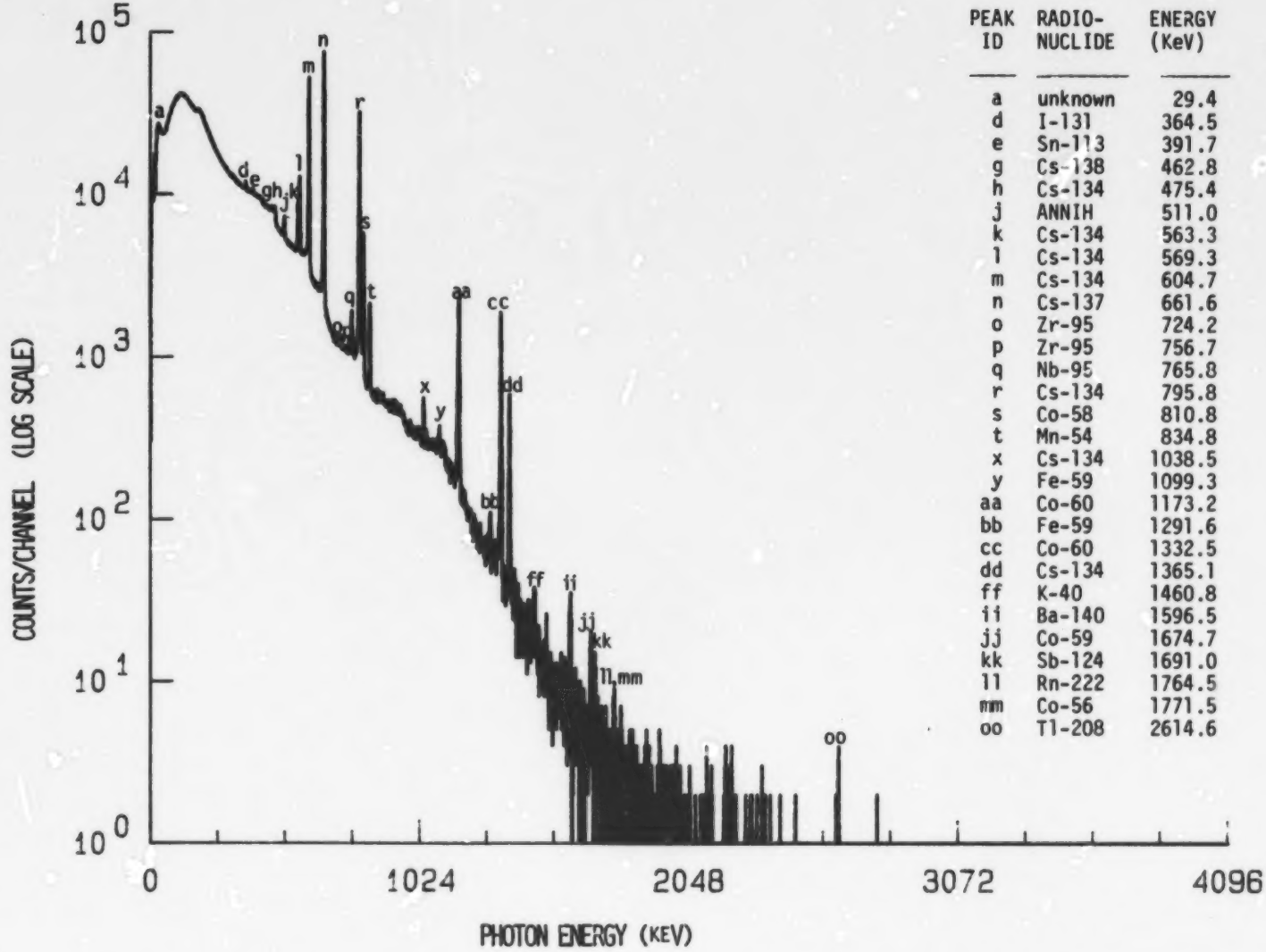
41



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)						
a	Xe-133	81.0	r	Nb-95	765.9	jj	Cs-134	1365.1
b	Kr-88	196.3	s	I-132	772.6	kk	Na-24	1368.5
c	Xe-135	249.8	t	Cs-134	795.8	ll	Cs-138	1435.9
d	Cr-51	320.1	u	Cs-134	801.8	mm	I-135	1457.6
e	Cd-115	336.3	v	Co-58	810.8	nn	K-40	1461.5
f	I-131	364.5	w	Kr-88	834.8	oo	Ni-65	1481.8
g	Cr-51	391.8	x	Kr-88	898.0	pp	Kr-88	1529.8
h	Cs-134	475.4	y	Cs-138	1009.8	qq	Ba-140	1596.5
i	ANNIH	511.0	z	Cs-134	1038.5	rr	I-135	1678.0
j	I-133	529.9	aa	I-135	1038.8	ss	Sb-124	1691.0
k	Cs-134	563.3	bb	Fe-59	1099.3	tt	Xe-138	1767.6
l	Cs-134	569.3	cc	I-135	1131.5	uu	Kr-88	1836.0
m	Cs-134	604.7	dd	Cs-134	1167.9	vv	Kr-88	2029.8
n	Cs-137	661.6	ee	Co-60	1173.2	ww	Kr-88	2035.4
o	I-132	667.7	ff	I-133	1236.4	xx	Kr-88	2195.8
p	Zr-95	724.2	gg	I-135	1260.4	yy	Kr-88	2392.1
q	Zr-95	756.7	hh	Fe-59	1291.6	zz	Kr-88	2677.9
			ii	Co-60	1332.5	aaa	Na-24	2753.9

FIGURE 12. Gamma Spectral Data - Inside Primary Pipe Penetration Room (GSM13) - During Routine Operations

BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
d	I-131	364.5
e	Sn-113	391.7
g	Cs-138	462.8
h	Cs-134	475.4
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
x	Cs-134	1038.5
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ff	K-40	1460.8
ii	Ba-140	1596.5
jj	Co-59	1674.7
kk	Sb-124	1691.0
ll	Rn-222	1764.5
mm	Co-56	1771.5
oo	Tl-208	2614.6

FIGURE 13. Gamma Spectral Data - Inside Pipe Penetration Room (POM06) - During An Outage

43

The data from outage time t=0 to t=80 days suggest that the estimated relative contributions of the activation products and the fission products to the total exposure rate appear to remain nearly constant over the time period studied. As the short-lived radionuclides decay to negligible levels, the longer-lived nuclides, such as Cs-137, Co-60, and Cs-134, dominate the radionuclides present and maintain a nearly constant ratio of activation product to fission product.

The calculations of radionuclides present and their contribution to the exposure rates at 80 days after shutdown can be compared directly to the measurements obtained on approximately the eighty-first day of the outage. The analysis of the gamma spectra obtained during routine operation's, on which decay calculations are based as is presented on page D.8. Analyses of the actual measurements obtained during the outage, but without the apparent low-energy x ray contribution, is presented on page D.9.

The apparent low-energy photon contribution was ignored for this comparison because the extrapolation of the data after plant shutdown was made based on radionuclides present and gamma peaks identified in the spectrum taken during routine operations measurement. The relative contributions of activation products and fission products to the total exposure rate compare favorably between the calculated estimates and the field observations. The total exposure rate calculated for the estimated and the measured values are different; the estimated exposure rate (3.0 mR/h) is greater than the measured exposure rate (1.7 mR/hr). This difference may result from the geometric correction factor, which was the established through the analysis of the exposure rate present during routine operations and was kept constant for all calculations. Although different radiation detectors were used for the two measurements, the analysis should have eliminated any differences resulting from use of the two detectors.

If the relative contributions of the activation products and fission products do remain constant during an outage period, any fuel cladding failure events would remain as significant to radiation exposure rates and occupational radiation exposures during the outage as during routine operations. The significance of fuel cladding failures during routine operations has been documented in NUREG/CR-4485 Vol. 1 to be more than a five-fold increase in radiation exposure rates in specific areas of the plant.

CONCLUSIONS

Based on the data collected during the refueling and maintenance outage of an 1100 MWe PWR nuclear power plant that had been operating with fuel having fuel cladding failure characterized as 0.21 percent, the following conclusions can be drawn:

- The data indicate that the relative contributions of the activation products and the fission products to the total exposure rate is nearly constant over the duration of the outage. This constancy is due to the significant contribution to the exposure rates of the longer-lived isotopes of cesium (a fission product) and cobalt (an activation product).

- Because the relative contributions of the activation and fission products remain nearly constant over the duration of outage, fuel cladding failures are as significant a contributor to radiation exposure rates and occupational radiation exposure during the outage as they are during routine operations. As reported in NUREG/CR-4485 Vol. 1, fuel cladding failures increased radiation exposure rates an estimated 540 percent in some areas of the plant during routine operations. Consequently, it can be concluded that fuel cladding failures resulted in many areas of the plant having radiation exposure rates significantly greater during the outage than would have been present following routine operations under normal fuel conditions; that is, without failed fuel.

- The apparent low-energy photopeaks appeared to contribute significantly to exposure rates in most locations, except those near steam generator #2. Unfortunately, the exact cause of the low-energy photopeaks has not been determined conclusively. The origin of this photopeak should be determined and warrants further investigation. The uncertainty regarding the origin of the photopeak should not, however, confuse the comparison of the contribution from fission products versus activation products to radiation exposure rates nor the conclusions derived from this critical comparison.

- Evaluation of multi-element beta thermoluminescent dosimeter radiation depth-dose data indicate that beta radiation is not a significant contributor to radiation dose rates in those locations within the plant that were evaluated. This small contribution is perhaps due to the fact that much of the beta radiation was attenuated within the piping system.

- Analyses of air samples taken inside containment for radioactive particulates and radioiodine indicate that at eighty-two days into the outage airborne radioactive materials and gases were not a significant contributor to occupational exposures at the locations measured. Discussions with plant personnel indicated that airborne radioactive materials and gases may have been at sufficiently high concentrations to impact occupational exposures inside containment at the beginning of the outage if workers had not utilized respiratory protection.

- Data collected at locations associated with refueling activities and spent fuel storage indicate that Co-58 is a principal contributor to exposure rates in these locations of the plant. The primary means of producing Co-58 is by activation of Ni-58 through a neutron-proton reaction. The Co-58 may be present because nickel or inconel alloys are used in components or piping. Activation products such as Co-58 are transported from the primary system to the spent fuel storage area on the fuel assemblies.

BIBLIOGRAPHY

Cember, H. 1969. Introduction to Health Physics. Pergamon Press, New York, New York.

International Commission on Radiological Protection (ICRP). 1983. Radionuclide Transformations. ICRP Publication 38, Pergamon Press, Elmsford, New York.

Meek, M. E., and B. F. Rider. 1972. Compilation of Fission Product Yields, Vallecitos Nuclear Center - 1972. NEDO-12154. General Electric Company, Pleasonton, California.

Moeller, M. P., G. A. Stoetzel, and L. H. Munson. 1985. Historical Summary of Occupational Radiation Exposure Experience in U.S. Commercial Nuclear Power Plants. NUREG/CR-4160, PNL-5404, Pacific Northwest Laboratory, Richland, Washington.

Moeller, M. P., G. F. Martin and D. L. Haggard. 1986. The Impact of Fuel Cladding Failure Events on Occupational Radiation Exposures at Nuclear Power Plants. Case Study: PWR During Routine Operations. NUREG/CR-4485 Vol. 1, PNL-5606, Pacific Northwest Laboratory, Richland, Washington.

Reus, U., W. Westmeier, and I. Warnecke. 1979. Gamma-Ray Catalog. GSI 79-2, Inst. for Nuclear Chemistry, University of Marburg, Lahnberge, West-Germany.

Roberson, P. L., G. W. R. Endres, R. A. Fox, D. L. Haggard, K. L. Holbrook, and L. A. Rathbun. 1984. Spectral and Dosimetric Measurements of Photon Fields at Commercial Nuclear Sites. NUREG/CR-3569, PNL-3585, Pacific Northwest Laboratory, Richland, Washington.

Scherpelz, R. I., G. W. R. Endres and L. A. Rathbun. 1983. The Use of Multielement Beta Dosimeters Measuring Dose Rates in the TMI-2 Containment Building. GEND-033, PNL-4714, Pacific Northwest Laboratory, Richland, Washington.

Shleien, B., and M. S. Terpilak (eds). 1984. The Health Physics and Radiological Health Handbook. Nucleon Lectern Associates, Olney, MD.

APPENDIX A

RADIONUCLIDE PRODUCTION

To discuss the impact of fuel cladding failures on occupational radiation exposure, it is useful to review the composition and means of production of the liquid and gaseous source terms encountered during the outage in the plant. In NUREG/CR-4485 Vol. 1, a review of the radionuclide composition of the liquid and gaseous source terms present during routine operations within a reactor system appeared as an appendix. The information discussed in this appendix is intended to provide more information on the principle means of production of the dominant radionuclides.

A review of the source terms encountered during an outage must include a discussion of how the source term changes with time after shutdown. As specific radionuclides decay according to their half-lives, the composition of the source term changes. Shorter lived radionuclides such as Kr-88, which was present during routine operations, decays with a half-life of 2.8 hours. Consequently, less than one day after shutdown, the impact of this radionuclide on occupational exposure is removed. Longer-lived radionuclides, such as Cs-134, Co-60, and Cs-137 with half-lives of 2.0, 5.2 and 30.0 years, decay much more slowly and, therefore, remain as part of the source term for an extended period of time. The source term from spent fuel at different time periods after shutdown is present in Table A.1.

Activation products and fission products comprise a significant amount of source term. The principal means of production of activation products is through neutron bombardment of the reactor's construction materials and the elements in the reactor's cooling water. The reactions of many significant activation products is presented in Table A.2. The principal means of production of fission products include fission, subsequent radioactive decay, neutron activation of the nuclides formed in the fission process. Generally, each fission produces two fission products resulting in a total fission yield of 200 percent. The fission product yield for each radionuclide represents its share of the total fission yield. An example of a fission product produced by neutron activation of a nuclide formed in the fission process is Cs-134. The nuclide formed in the fission process is stable Cs-133, with a fission product yield of 6.6 percent. Through a neutron-gamma reaction, Cs-133 is activated to form Cs-134. Fission product yields and half-lives information for the significant fission products is presented in Table A.3.

TABLE A.1. Activity of Fission Products in Curies at Specified Times (T) After Removal From a Reactor That Has Operated at 1000 kW Energy for 1 Year

Fission product	$T = 0$[a]	$T = 100$ days	$T = 1$ year	$T = 5$ years
Kr-85	191	187	177	132
Rb-86	0.26	—	—	—
Sr-89	38,200	10,300	321	—
Sr-90	1,430	1,420	1,380	1,200
Y-90[b]	1,430	1,420	1,380	1,200
Y-91	48,900	14,500	577	—
Zr-95	49,200	17,000	1,000	—
Nb-95(90 H)[b]	687	152	15	—
Nb-95(35 D)[b]	48,200	28,700	2,140	—
Ru-103	30,900	5,920	74	—
Rh-103[b]	30,900	5,920	74	—
Ru-106	2,180	1,800	1,090	68
Rh-106[b]	2,180	1,800	1,090	68
Ag-111	151	—	—	—
Cd-115	5.9	1.2	—	—
Sn-117	84	0.7	—	—
Sn-119	<64	<48	<23	<0.4
Sn-123	9	5	1	—
Sn-125	101	0.1	—	—
Sb-125[b]	43	41	34	12
Te-125[b]	34	39	36	13
Sb-127	787	—	—	—
Te-127(90 D)[b]	260	123	16	—
Te-127(9.3 H)[b]	922	124	16	—
Te-129(32 D)	1,590	182	0.6	—
Te-129(70 M)[b]	1,590	182	0.6	—
I-131	25,200	4	—	—
Xe-131[b]	252	1.5	—	—
Te-132	36,900	—	—	—
I-132[b]	36,900	—	—	—
Xe-133	55,300	—	—	—
Cs-136	52	0.3	—	—
Cs-137	1,080	1,070	1,060	970
Ba-137[b]	1,030	1,020	1,010	920
Ba-140	51,700	230	—	—
La-140[b]	51,700	265	—	—
Ce-141	47,800	4,740	10	—
Pr-143	45,300	288	—	—
Ce-144	26,700	20,800	10,700	268
Pr-144[b]	26,700	20,800	10,700	268
Nd-147	21,800	40	—	—
Pm-147[b]	4,900	4,800	3,950	1,360
Sm-151	37	37	36.6	35
Eu-155	74	67	52	13
Eu-156	109	1.2	—	—
Total	693,573	144,130	36,964	6,527

[a] Calculated using fission product yields. [b] Daughter product.
From *Radiological Health Handbook*, 1960.

(1) Reprinted with permission from Introduction to Health Physics, by Herman Cember, Volume 105 of the International Series of Monographs in Nuclear Energy, Copyright 1969, Pergamon Press, Ltd.

TABLE A.2. Activation Products

Nuclide	Half-Life	Mode of Production	Major Source
Cr-51	27.8d	$^{50}Cr\ (n,\gamma)\ ^{51}Cr$	Steel and nickel alloys
Mn-54	303d	$^{54}Fe\ (n,p)\ ^{54}Mn$	Steel and nickel alloys
Co-56	77.3d	$^{56}Fe\ (p,n)\ ^{56}Co$	Steel and nickel alloys
Fe-59	45d	$^{58}Fe\ (n,\gamma)\ ^{59}Fe$	Steel and nickel alloys
Co-58	71d	$^{58}Ni\ (n,p)\ ^{58}Co$	Steel and nickel alloys
Co-60	5.24	$^{59}Co\ (n,\gamma)\ ^{60}Co$	Stellite steel
Zr-65	250d	$^{64}Zn\ (n,\gamma)\ ^{65}Zn$	Piping
Ag-110m	255d	$^{109}Ag\ (n,\gamma)\ ^{110m}Ag$	Control rods
Sr-113	115d	$^{112}Sn\ (n,\gamma)\ ^{113}Sn$	Piping
		$^{113}In\ (p,n)\ ^{113}Sn$	Piping
Sb-124	60.4d	$^{123}Sb\ (n,\gamma)\ ^{124}Sb$	Piping
Sb-125	2.71y	$^{124}Sb\ (n,\gamma)\ ^{125}Sb$	Piping
		$^{125}Sn \rightarrow {}^{125}Sb+B^-$	Piping
Eu-148	54.5d	$^{148}Sm\ (p,n)\ ^{148}Eu$	Steel Alloy, piping, neutron absorbing material

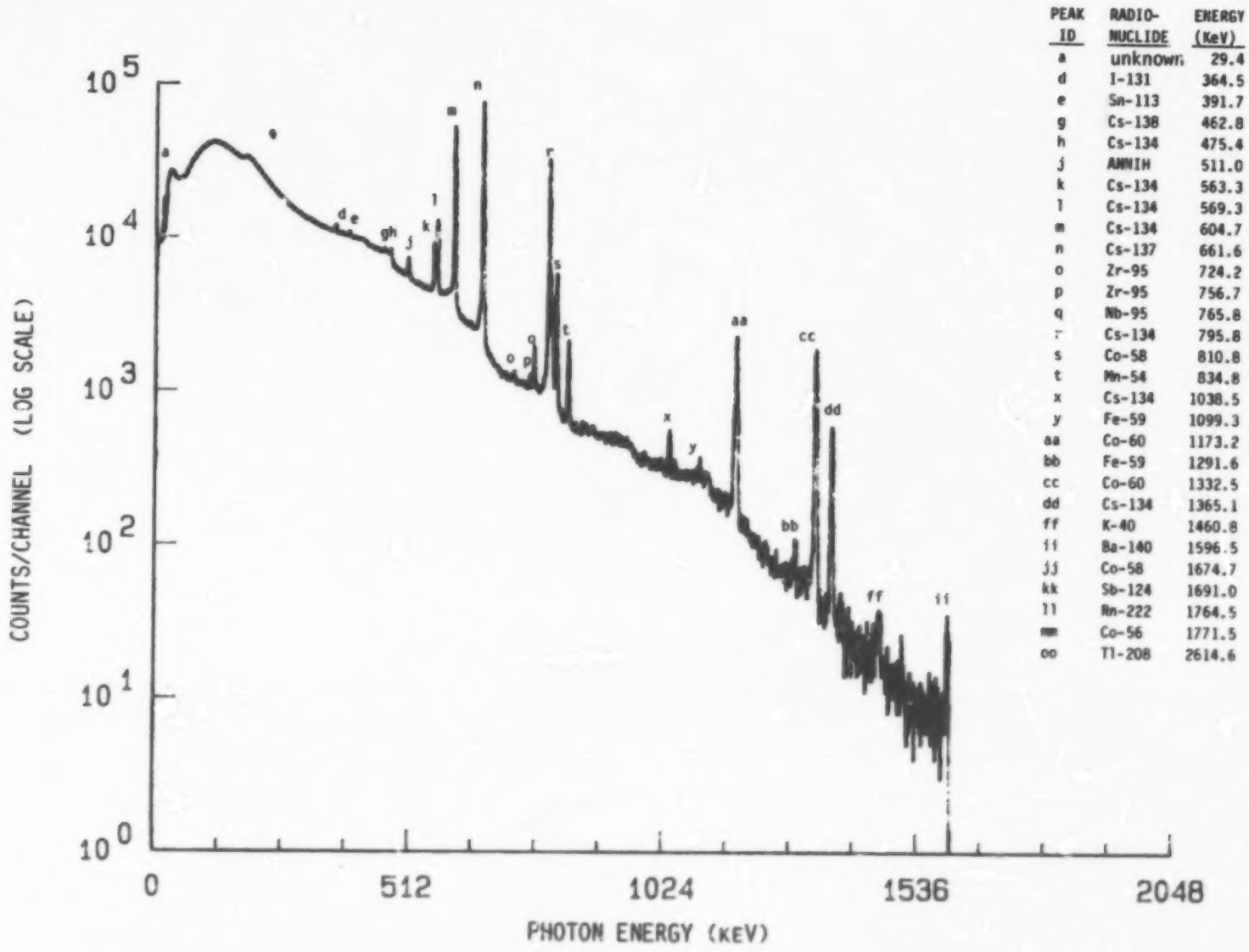
TABLE A.3. Fission Products

Nuclide	Half-Life	Yield (%)	Comments
Nb-95	35d	6.27	
Zr-95	65.5d	6.27	
Ru-103	39.5d	3.22	
Ru-106	368d	0.97	
Te-127m	109d	0.06	
I-131	8.05d	3.21	
Xe-133	5.29d	6.60	
Cs-133	Stable	6.60	
Cs-134	2.5y	<<1	$^{133}Cs\ (n,\gamma)\ ^{134}Cs$
Cs-138	32 min	6.50	
Ba-140	12.8d	5.90	
Ce-141	32.0d	5.96	
Ce-144	284d	3.54	
Eu-154	8.5y	0.0003	
Eu-155	4.68y	0.066	

APPENDIX B

GAMMA SPECTRAL ANALYSES

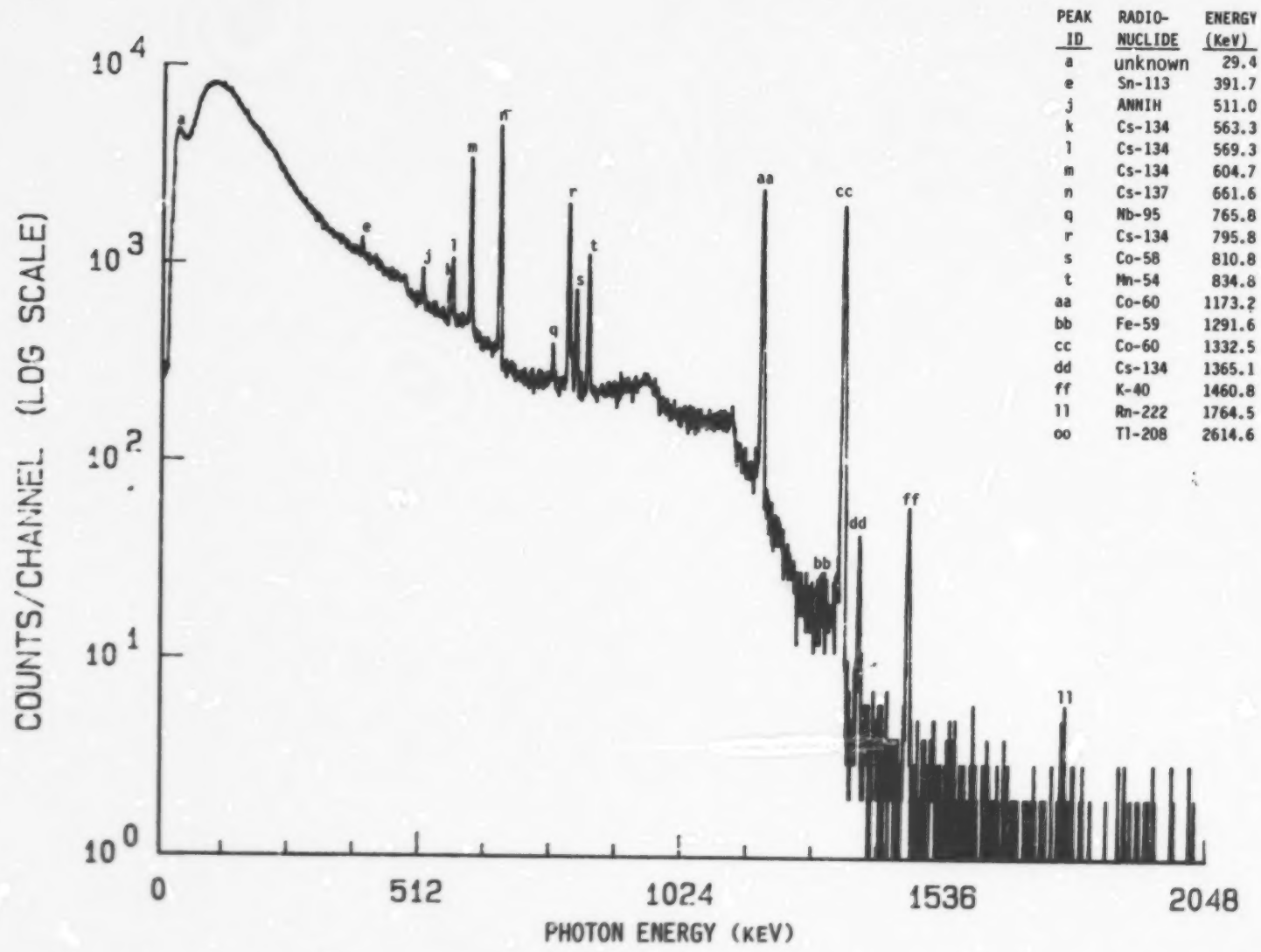
The gamma spectroscopy data for the 19 measurements made in the radwaste, containment, and pipe penetration building are presented in Figures B.1 through B.19. The photon peaks have been identified by the characteristic energy of the contributing radionuclide. In the analysis of each gamma spectrum, the peaks are labeled in the same manner to allow cross-referencing. The gamma spectra are presented in the same order as described in the results section. This order covers the plant areas of the chemical and volume control system, primary coolant system and components, and refueling activities and spent fuel storage.



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
d	I-131	364.5
e	Sn-113	391.7
g	Cs-138	462.8
h	Cs-134	475.4
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
x	Cs-134	1038.5
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ff	K-40	1460.8
ii	Ba-140	1596.5
jj	Co-58	1674.7
kk	Sb-124	1691.0
ll	Rn-222	1764.5
mm	Co-56	1771.5
oo	Tl-208	2614.6

FIGURE B.1. Gamma Spectra Obtained Inside Pipe Penetration (POM06)

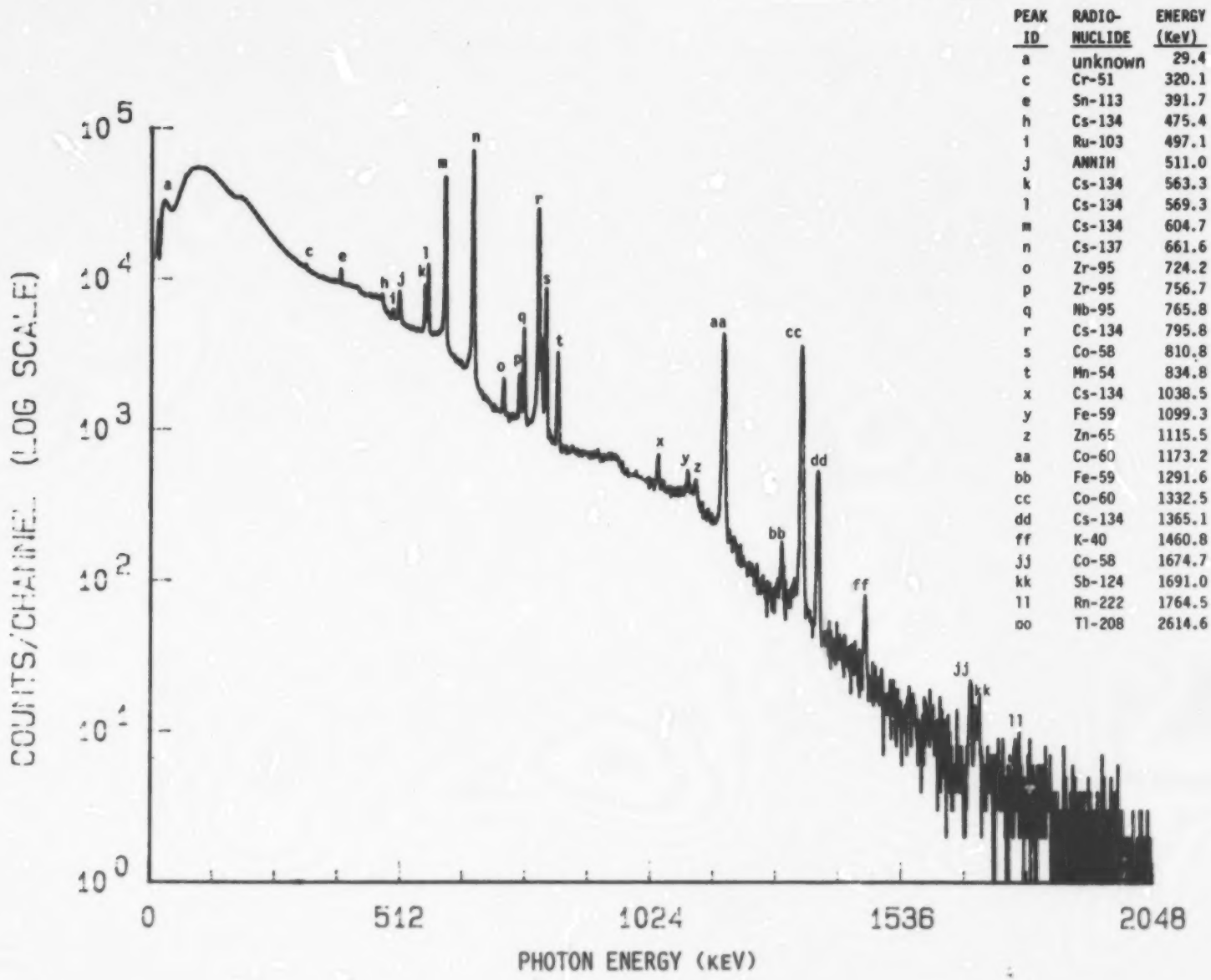
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
e	Sn-113	391.7
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ff	K-40	1460.8
ll	Rn-222	1764.5
oo	Tl-208	2614.6

FIGURE B.2. Gamma Spectra Obtained at the Purifying Ion-exchanger Valve Room/Inlet (POM01)

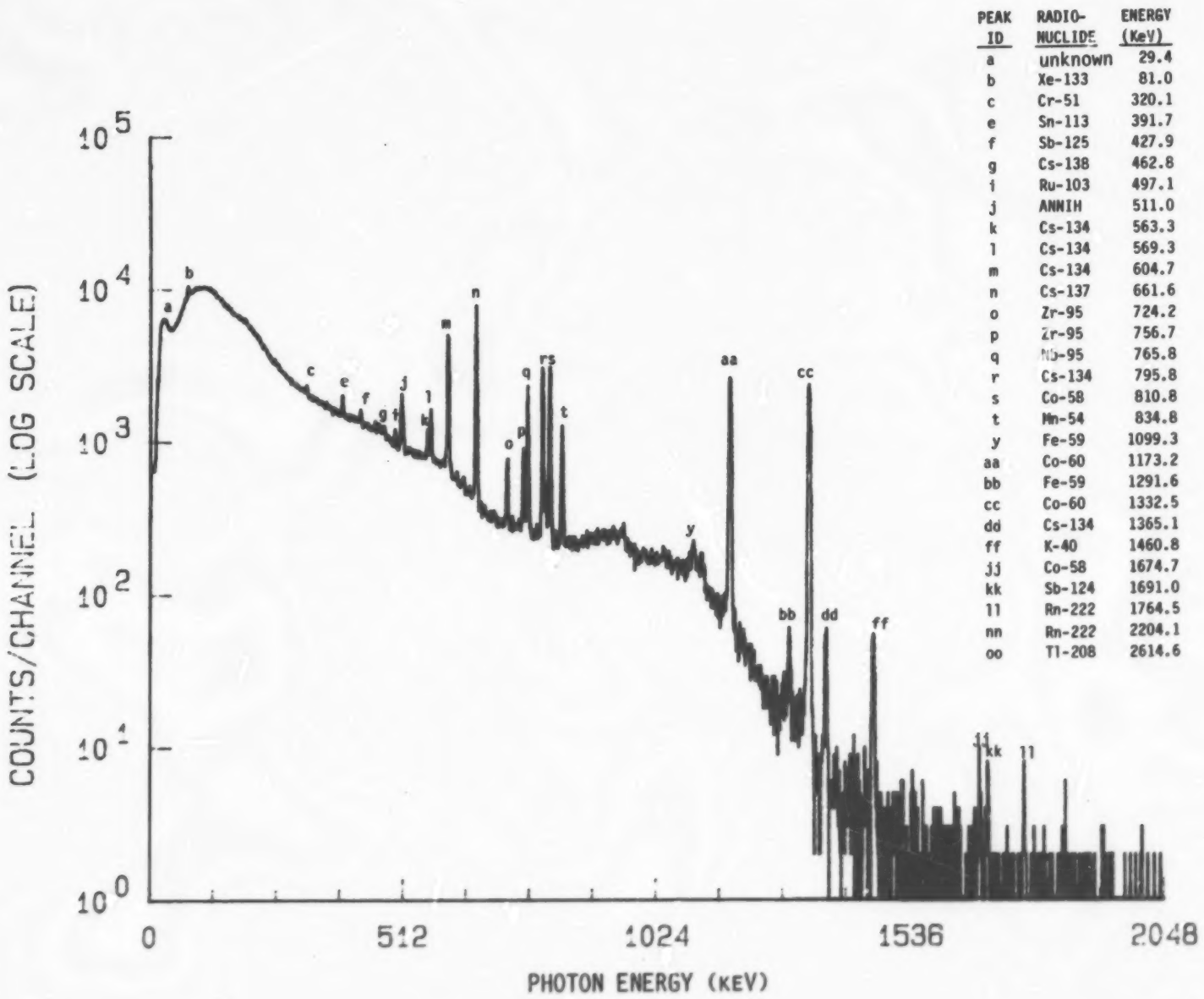
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
c	Cr-51	320.1
e	Sn-113	391.7
h	Cs-134	475.4
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
x	Cs-134	1038.5
y	Fe-59	1099.3
z	Zn-65	1115.5
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ff	K-40	1460.8
jj	Co-58	1674.7
kk	Sb-124	1691.0
ll	Rn-222	1764.5
oo	Tl-208	2614.6

FIGURE B.3. Gamma Spectra Obtained Inside Bypass Ion exchanger Valve Room/Outlet (POM04)

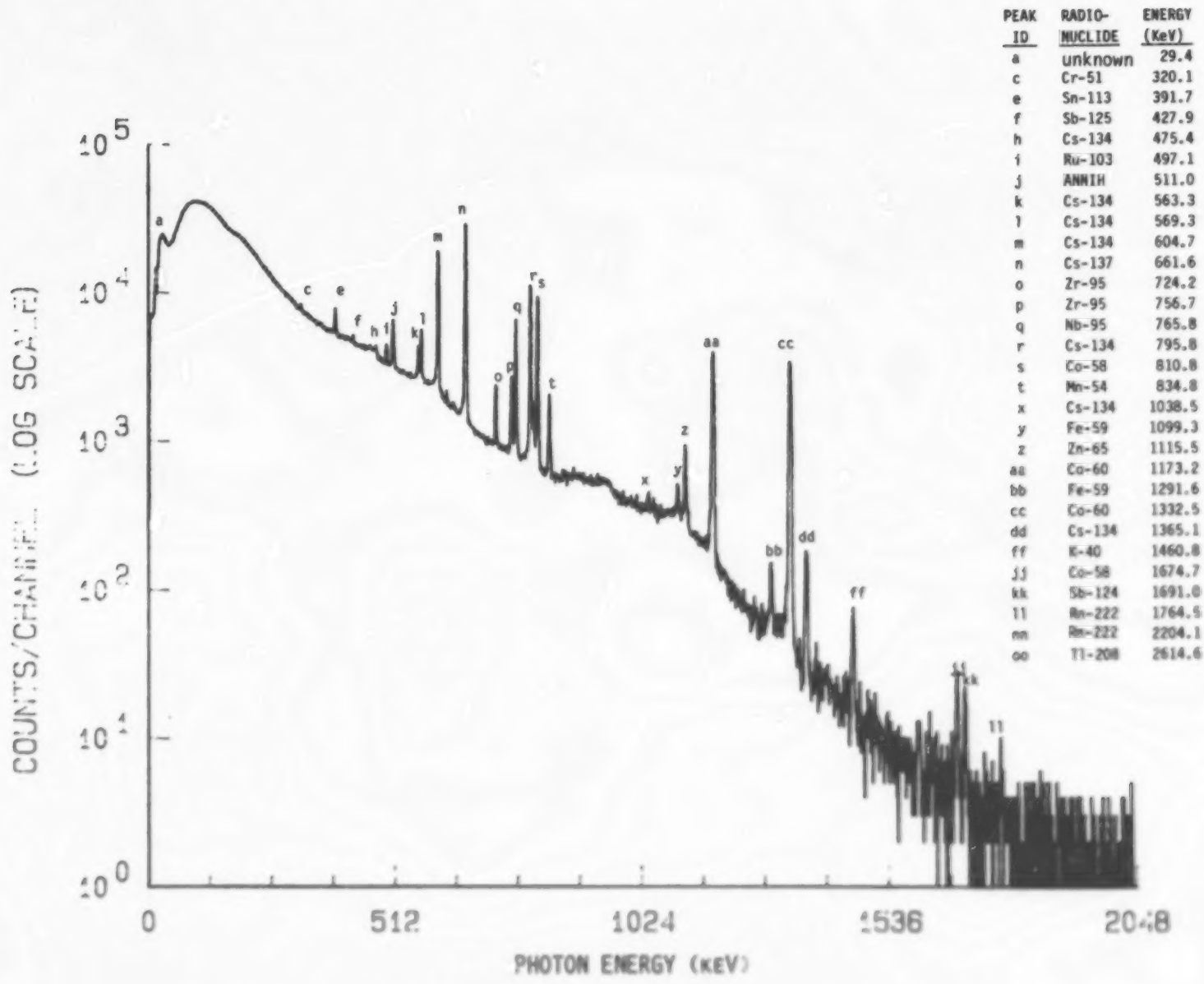
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
b	Xe-133	81.0
c	Cr-51	320.1
e	Sn-113	391.7
f	Sb-125	427.9
g	Cs-138	462.8
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ff	K-40	1460.8
jj	Co-58	1674.7
kk	Sb-124	1691.0
ll	Rn-222	1764.5
nn	Rn-222	2204.1
oo	Tl-208	2614.6

FIGURE B.4. Gamma Spectra Obtained at the Pressurizer Off-gass Line (POM02)

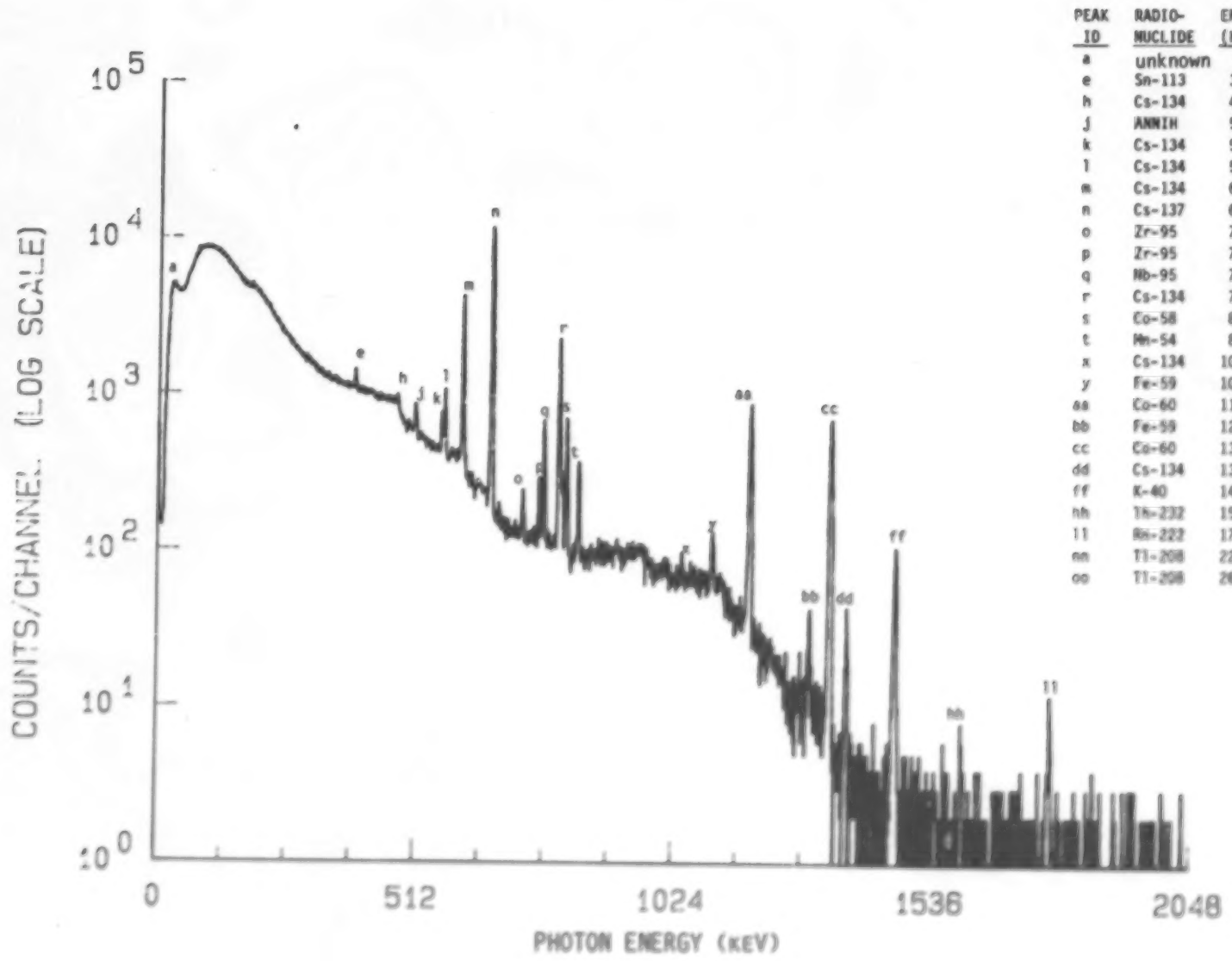
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
c	Cr-51	320.1
e	Sn-113	391.7
f	Sb-125	427.9
h	Cs-134	475.4
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
x	Cs-134	1038.5
y	Fe-59	1099.3
z	Zn-65	1115.5
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ff	K-40	1460.8
jj	Co-58	1674.7
kk	Sb-124	1691.0
ll	Rn-222	1764.5
mm	Rn-222	2204.1
oo	Tl-208	2614.6

FIGURE B.5. Gamma Spectra Obtained Inside Pressurizer Level Control Valve Room (POM03)

BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
e	Sn-113	391.7
h	Cs-134	475.4
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
x	Cs-134	1038.5
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ff	K-40	1460.8
hh	Th-232	1587.9
ii	Rn-222	1764.5
nn	Tl-208	2204.1
oo	Tl-208	2614.6

FIGURE B.6. Gamma Spectra Obtained Inside Charging Pump Room (POMOS)

BEST COPY AVAILABLE

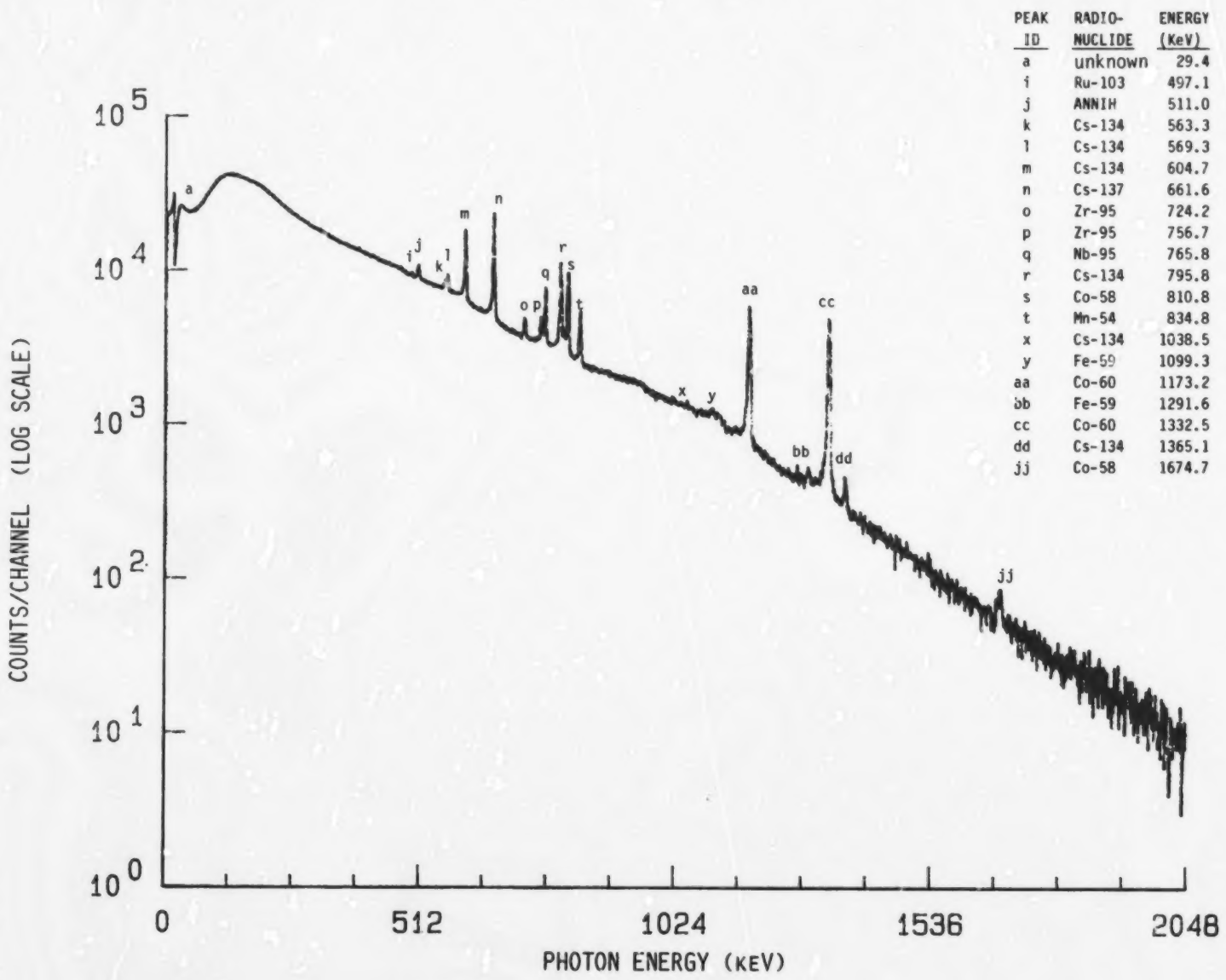
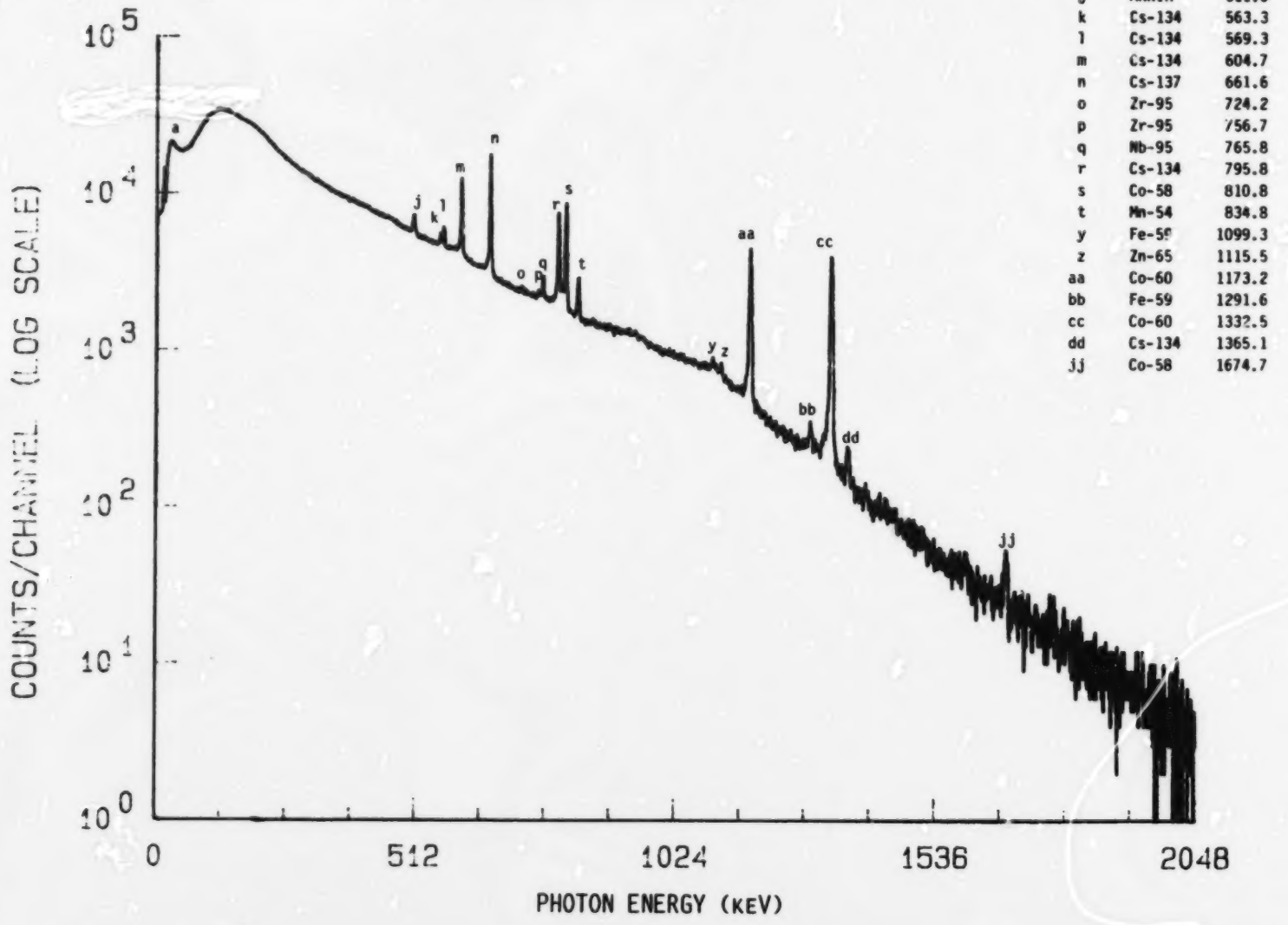


FIGURE B.7. Gamma Spectra Obtained at the Pressurizer Surge Line (POM16)

BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
y	Fe-59	1099.3
z	Zn-65	1115.5
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
jj	Co-58	1674.7

FIGURE B.8. Gamma Spectra Obtained at the Hot Leg of Steam Generator #2 (POM13)

BEST COPY AVAILABLE

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
g	Cs-138	462.8
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1

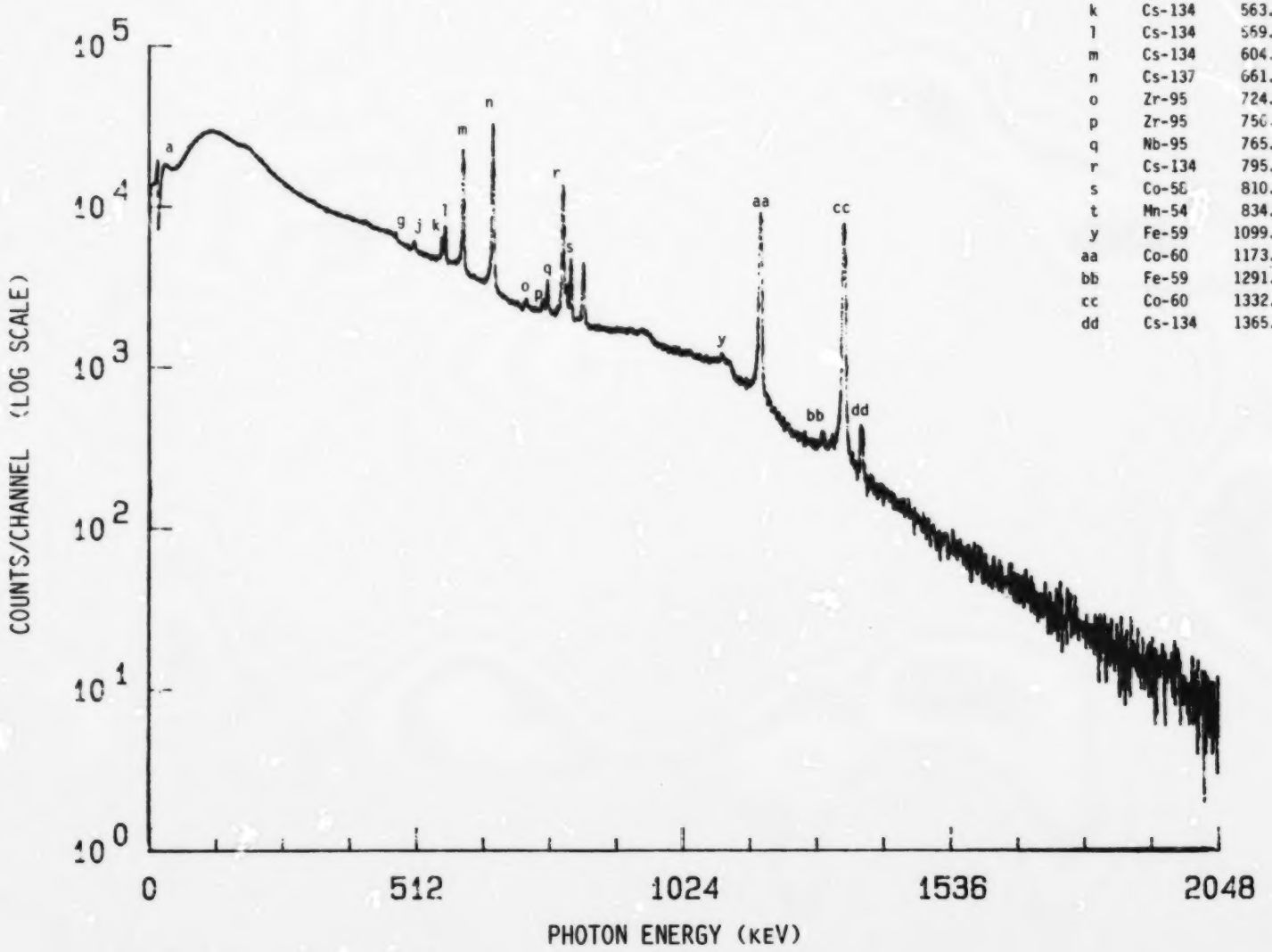
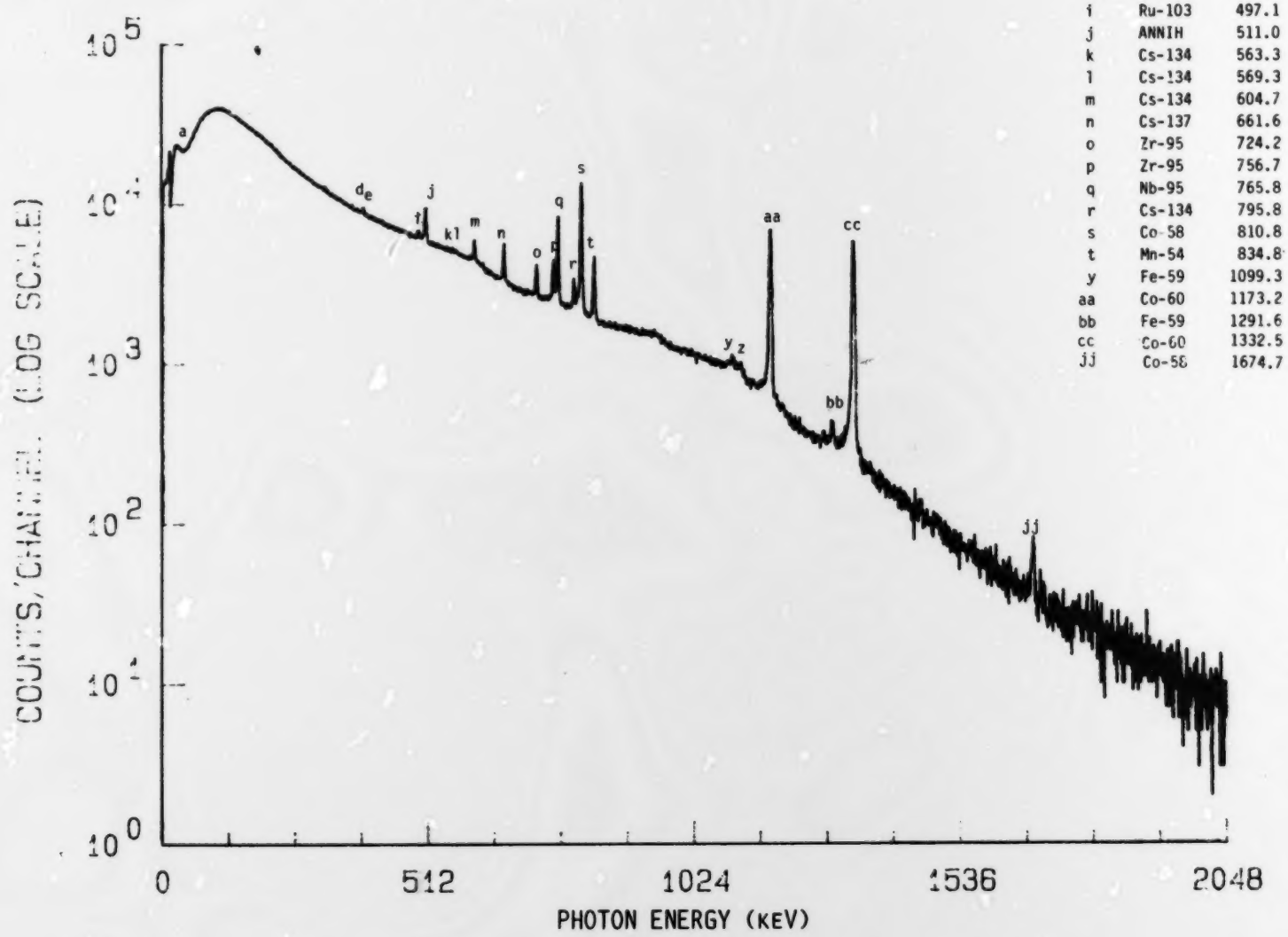


FIGURE B.9. Gamma Spectra Obtaind at the Value Near Steam Generator #2 (POM08)

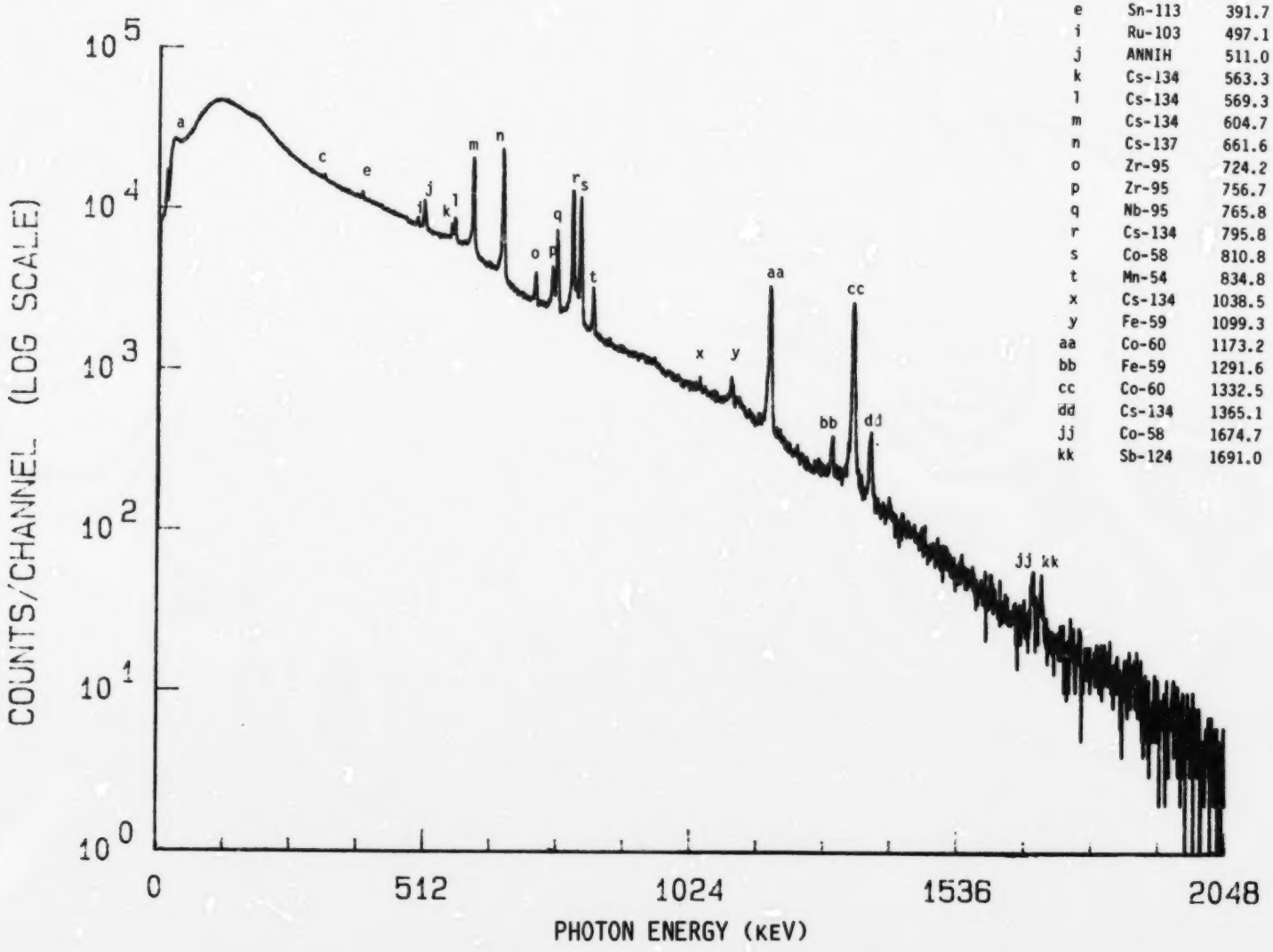
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
d	I-131	364.5
e	Sn-113	391.7
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
jj	Co-58	1674.7

FIGURE B.10. Gamma Spectra Obtained at the Letdown Valve Near Steam Generator #1 (POM10)

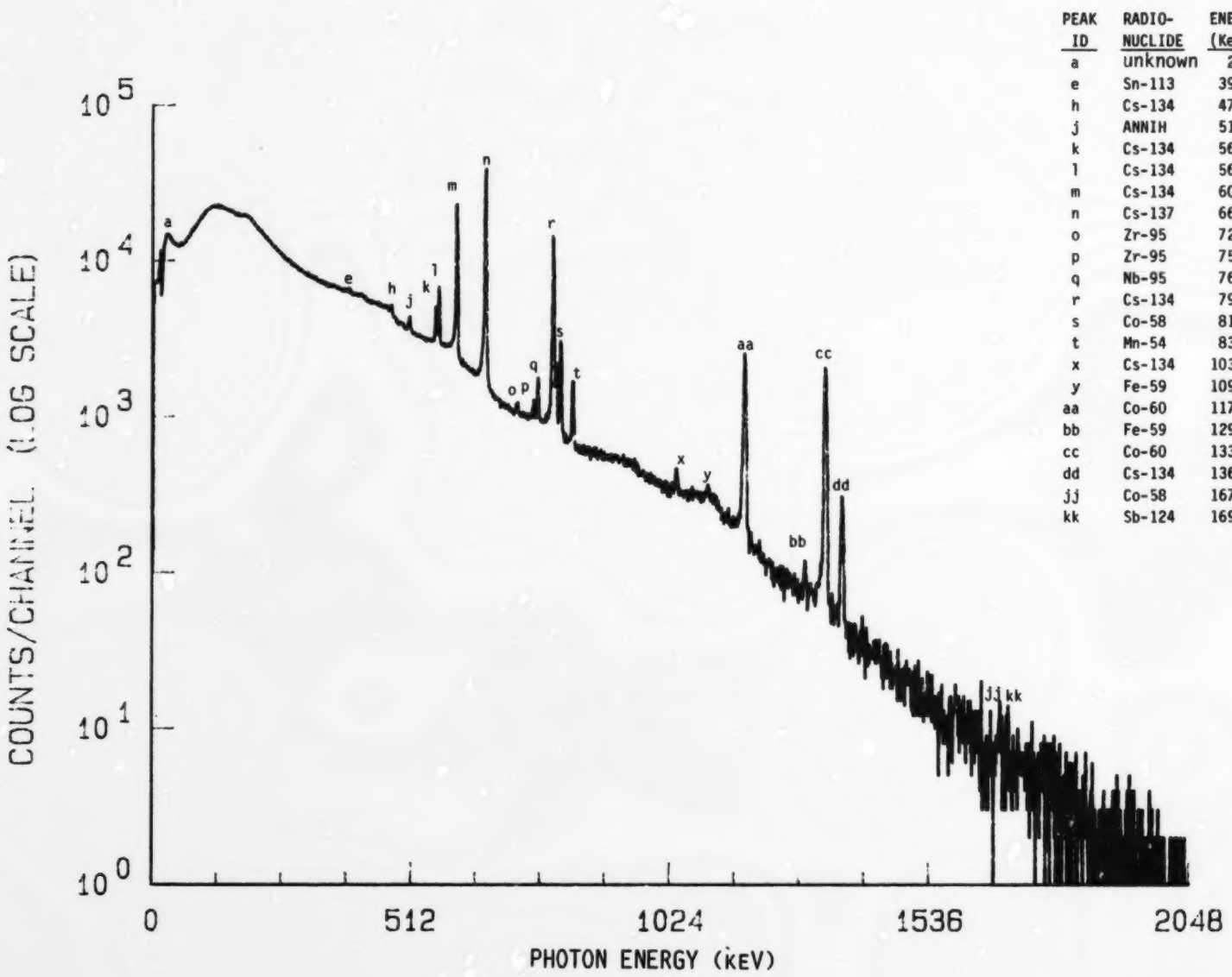
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
c	Cr-51	320.1
e	Sn-113	391.7
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
x	Cs-134	1038.5
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
jj	Co-58	1674.7
kk	Sb-124	1691.0

FIGURE B.11. Gamma Spectra Obtained at the Shutdown Cooling Line - 17'6" - Level (POM09)

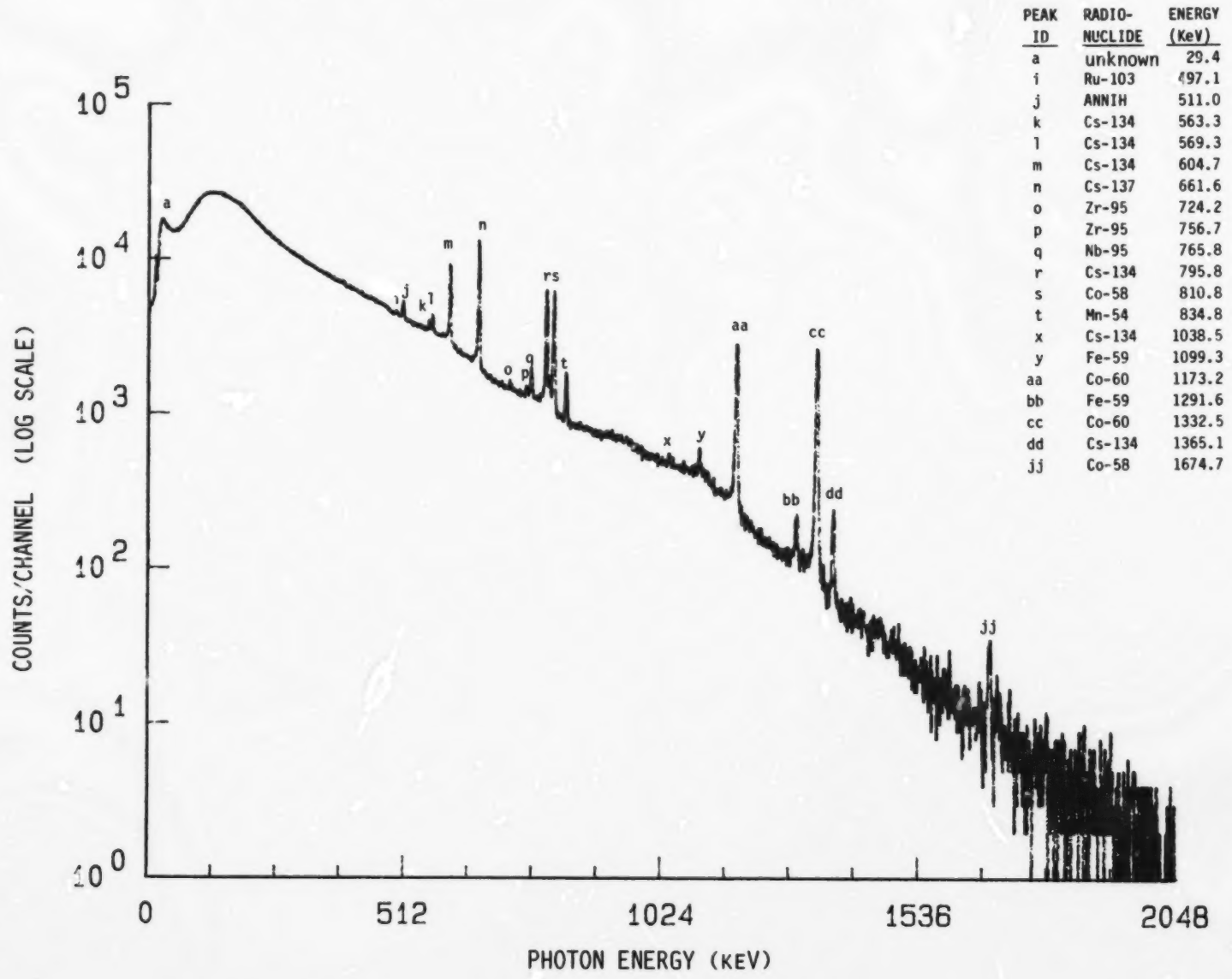
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
e	Sn-113	391.7
h	Cs-134	475.4
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
x	Cs-134	1038.5
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
jj	Co-58	1674.7
kk	Sb-124	1691.0

FIGURE B.12. Gamma Spectra Obtained at the Shutdown Cooling Line - 30' - Level (POM12)

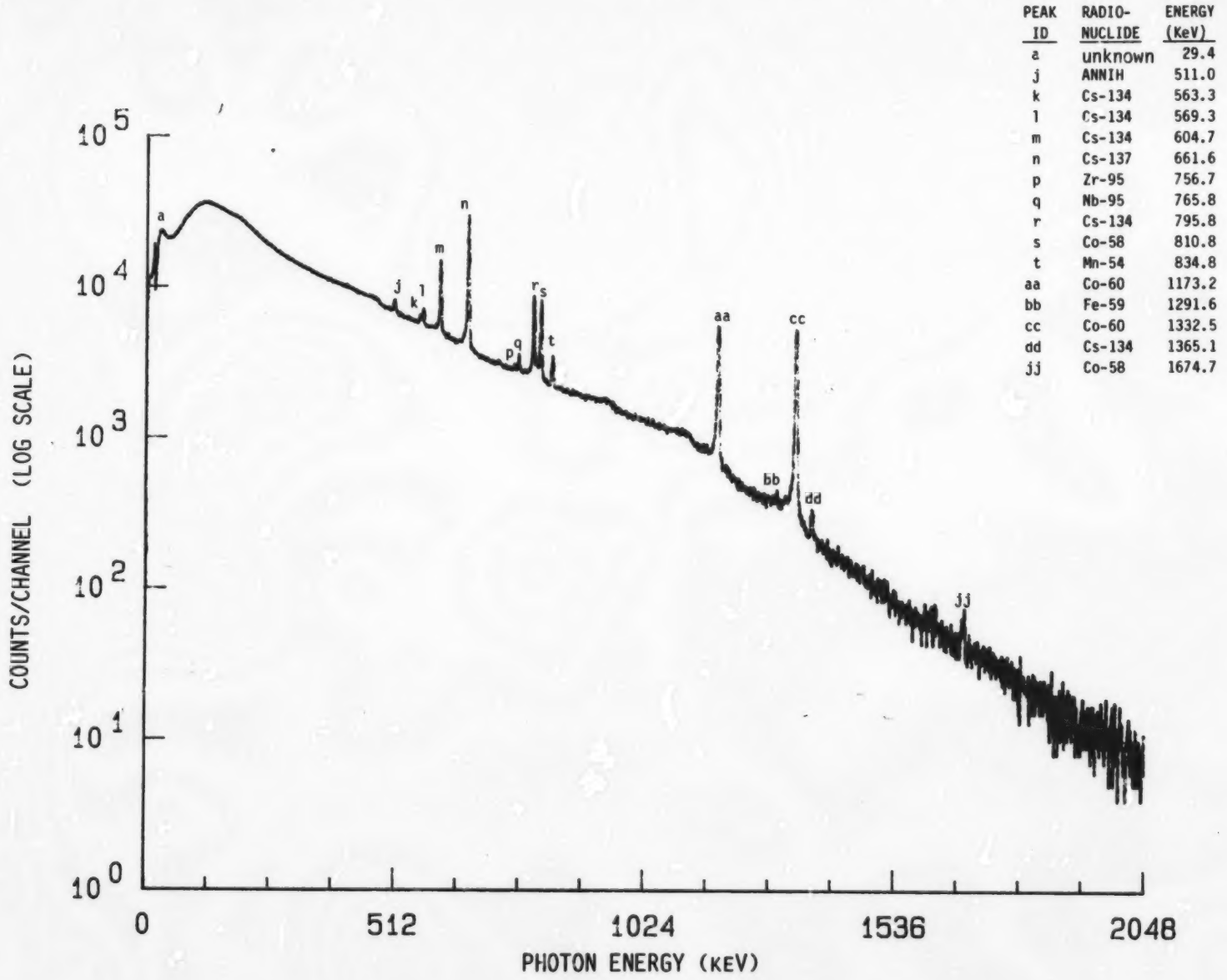
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
x	Cs-134	1038.5
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
jj	Co-58	1674.7

FIGURE B.13. Gamma Spectra Obtained at the Shutdown Cooling Line Upstream of POM14 Valve (POM15)

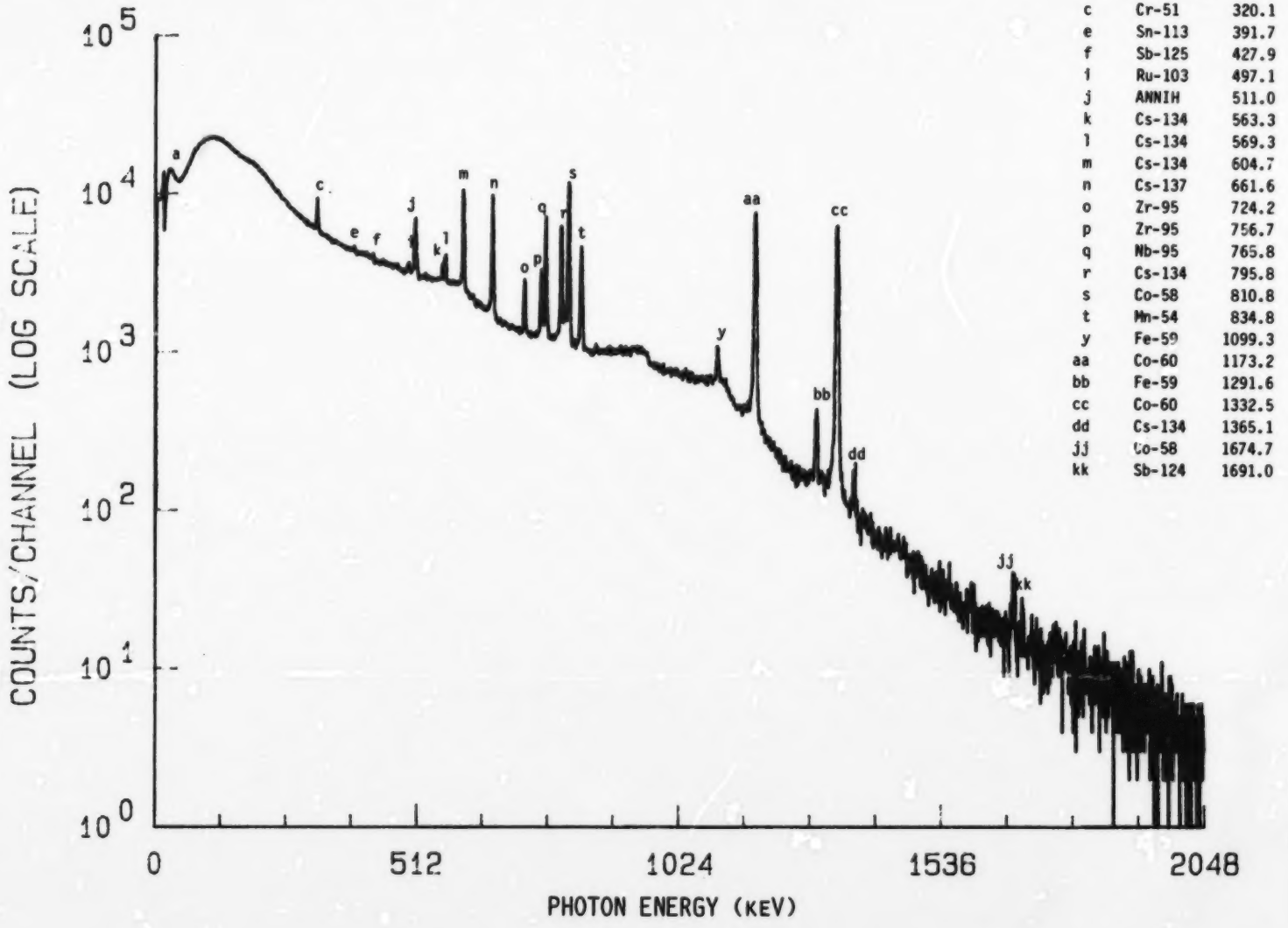
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
jj	Co-58	1674.7

FIGURE B.14. Gamma Spectra Obtained at the Shutdown Cooling Isolation Valve (POM14)

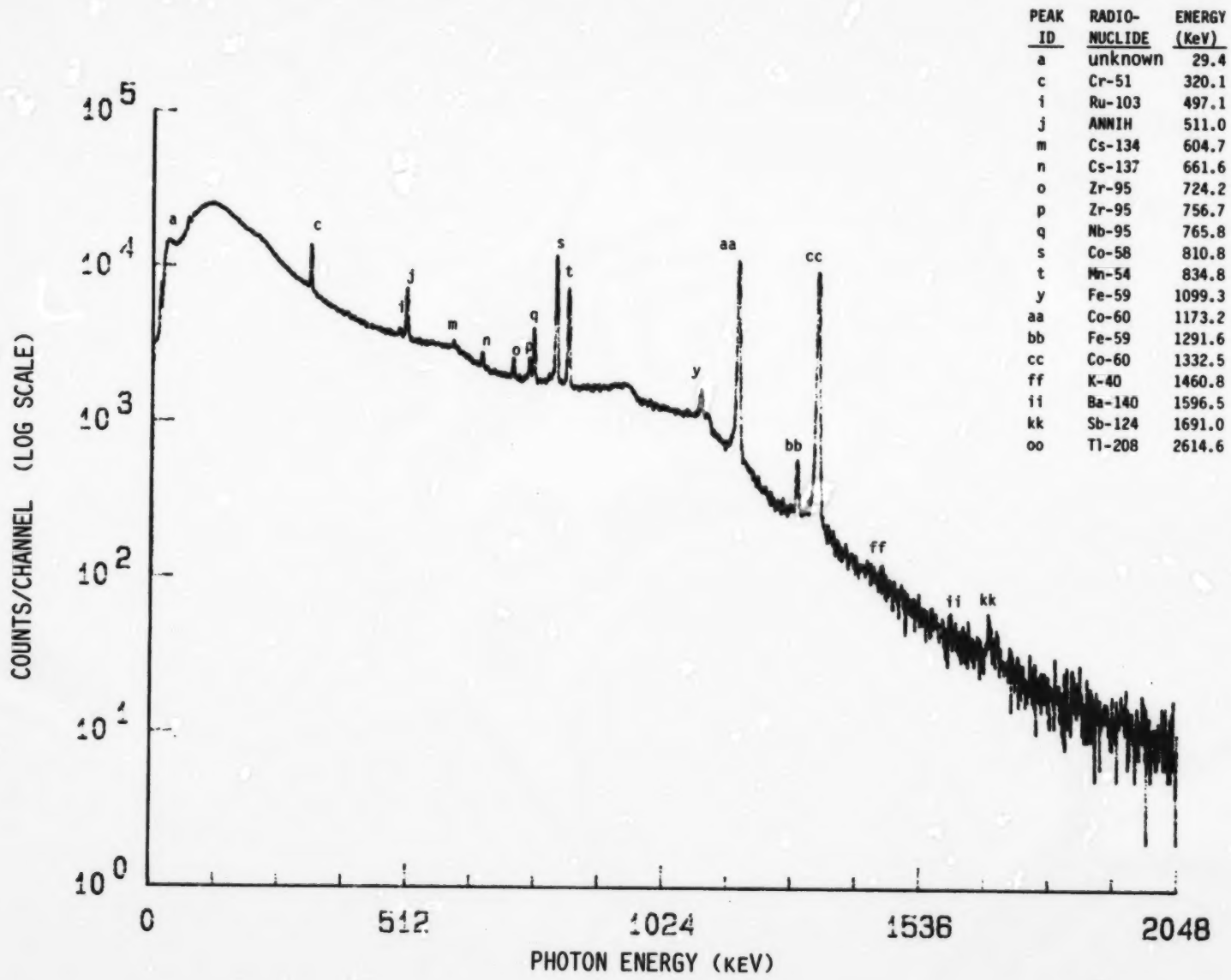
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
c	Cr-51	320.1
e	Sn-113	391.7
f	Sb-125	427.9
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
jj	Co-58	1674.7
kk	Sb-124	1691.0

FIGURE B.15. Gamma Spectra Obtained at the Cavity Drain Line Near Elevation Access Point (POM11)

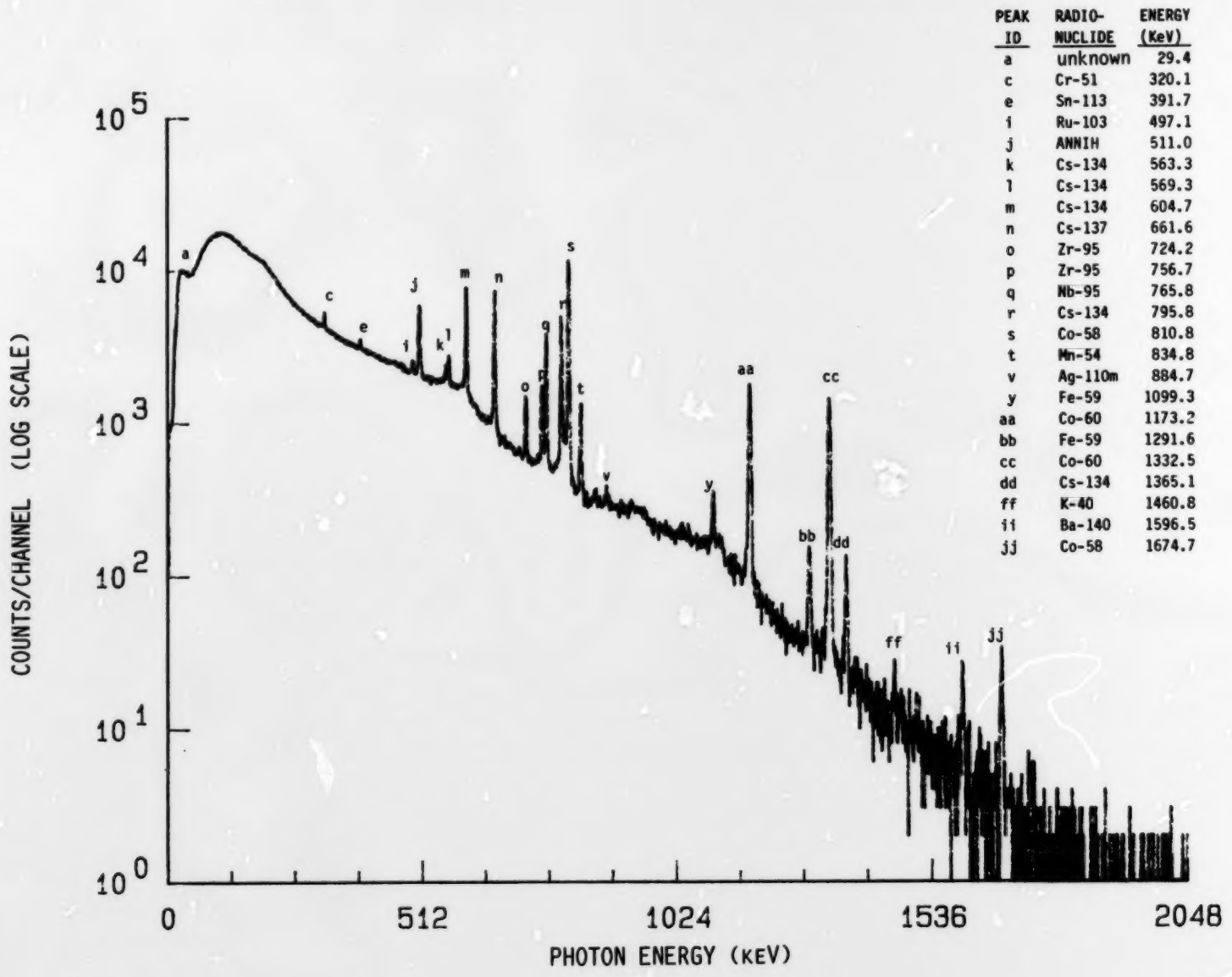
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
c	Cr-51	320.1
i	Ru-103	497.1
j	ANNIH	511.0
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
s	Co-58	810.8
t	Mn-54	834.8
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
ff	K-40	1460.8
ii	Ba-140	1596.5
kk	Sb-124	1691.0
oo	Tl-208	2614.6

FIGURE B.16. Gamma Spectra Obtained at the Main Cavity Drain Line (POM07)

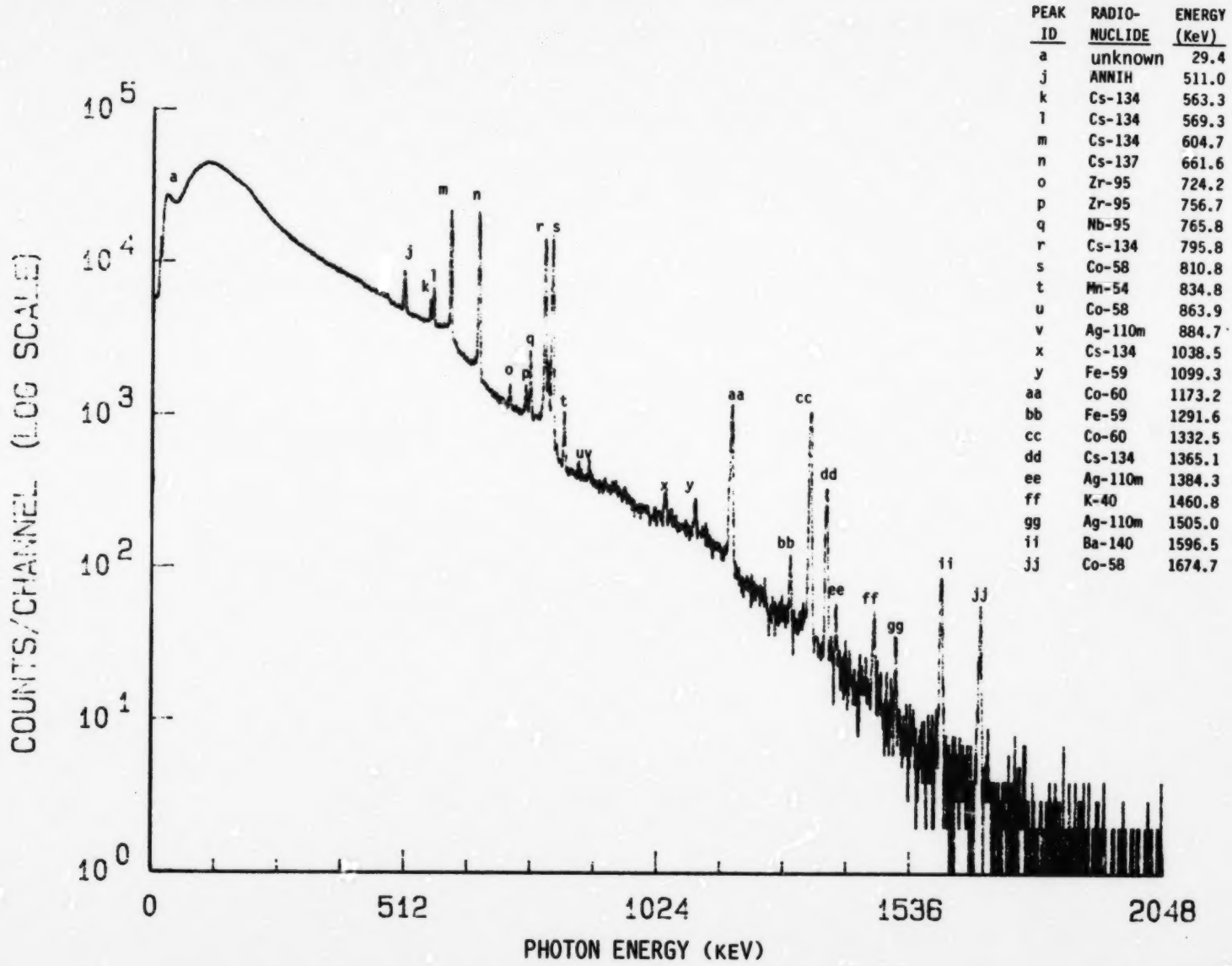
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
c	Cr-51	320.1
e	Sn-113	391.7
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
v	Ag-110m	884.7
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ff	K-40	1460.8
ii	Ba-140	1596.5
jj	Co-58	1674.7

FIGURE B.17. Gamma Spectra Obtained Inside Spent Fuel Pool Pump Room (POM17)

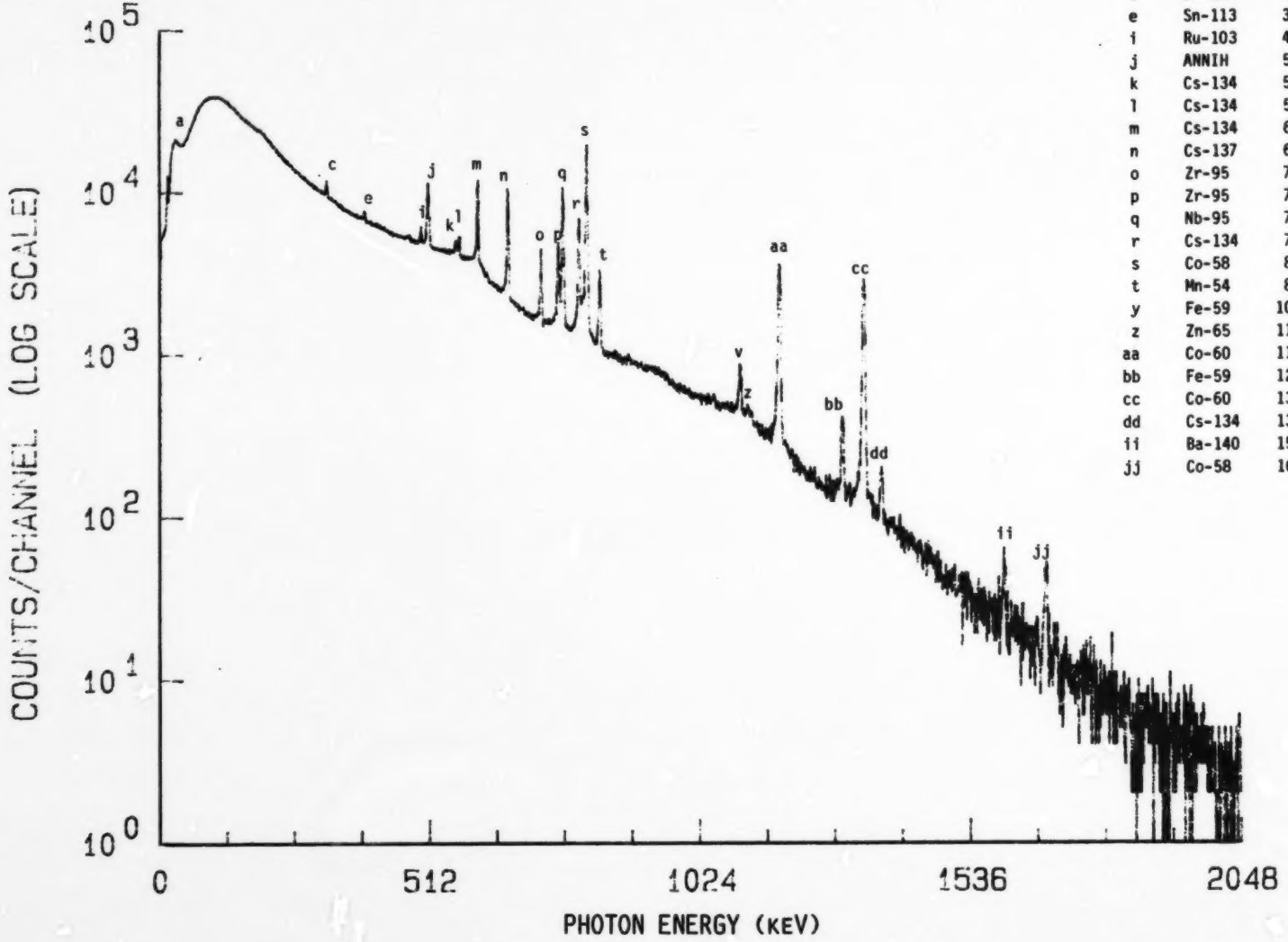
BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
u	Co-58	863.9
v	Ag-110m	884.7
x	Cs-134	1038.5
y	Fe-59	1099.3
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ee	Ag-110m	1384.3
ff	K-40	1460.8
gg	Ag-110m	1505.0
ii	Ba-140	1596.5
jj	Co-58	1674.7

FIGURE B.18. Gamma Spectra Obtained at the Fuel Pool Heat Exchanger (POM18)

BEST COPY AVAILABLE



PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)
a	unknown	29.4
c	Cr-51	320.1
e	Sn-113	391.7
i	Ru-103	497.1
j	ANNIH	511.0
k	Cs-134	563.3
l	Cs-134	569.3
m	Cs-134	604.7
n	Cs-137	661.6
o	Zr-95	724.2
p	Zr-95	756.7
q	Nb-95	765.8
r	Cs-134	795.8
s	Co-58	810.8
t	Mn-54	834.8
y	Fe-59	1099.3
z	Zn-65	1115.5
aa	Co-60	1173.2
bb	Fe-59	1291.6
cc	Co-60	1332.5
dd	Cs-134	1365.1
ii	Ba-140	1596.5
jj	Co-58	1674.7

FIGURE B.19. Gamma Spectra Obtained at the Fuel Pool Heal Exchanger - Main Line (POM19)

BEST COPY AVAILABLE

APPENDIX C

ATTRIBUTABLE EXPOSURE RATES DERIVED FROM THE GAMMA SPECTRAL DATA

The gamma spectroscopy data have been analyzed to determine the exposure rates attributable to individual radionuclides. With this information, it is possible to estimate the total exposure rate attributable to all of the activation products and to all of the fission products. This summation facilitates a determine of their collective potential impact on occupational radiation exposures. Radionuclides produced by means other than an activation or fission process are listed in the group labeled other. The analyses of the 19 gamma spectroscopy measurements are presented here. The analyses of the gamma spectra are presented in the same order as described in the results section. The analyses are presented according to the plant areas of the chemical and volume control system, primary coolant system and components, and refueling activities and spent fuel storage.

A note of caution is in order with regard to the analysis of the recurring photopeak spanning from approximately 26 to 33 keV in each of the spectra. Figures B.2 and B.6 appear to indicate that the low-energy photopeaks are not an artifact of the discriminator level. Unfortunately, the exact cause of the low-energy photopeaks has not been determined conclusively. Consequently, when reviewing the data analysis presented in this appendix, it must be remembered that there remains uncertainty with regard to the contribution of the low-energy photopeaks to exposure rate. This uncertainty should not, however, confuse the comparison of the contribution from fission products versus activation products to radiation exposure rates nor the conclusions derived from this critical comparison.

****

PLANT: UNIT B PENETRATION BUILDING

LOCATION: INSIDE PIPE PENETRATION ROOM

FILE NAME: POM06 GEOMETRIC CORRECTION FACTOR: 5.17E+03

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 22.3

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 1.70E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	132338.	4.14E-01	23.69
d	I-131	364.5	Fission	3367.	2.40E-03	0.14
e	Sn-113	391.7	Activation	1705.	1.39E-03	0.08
g	Cs-138	462.8	Fission	581.	6.33E-04	0.04
h	Cs-134	475.4	Fission	3130.	3.56E-03	0.20
j	ANNIH	511.0	Other	6881.	9.03E-03	0.52
k	Cs-134	563.3	Fission	16598.	2.57E-02	1.47
l	Cs-134	569.3	Fission	22736.	3.59E-02	2.06
m	Cs-134	604.7	Fission	157626.	2.76E-01	15.78
n	Cs-137	661.6	Fission	256608.	5.22E-01	29.86
o	Zr-95	724.2	Fission	629.	1.48E-03	0.08
p	Zr-95	756.7	Fission	860.	2.18E-03	0.12
q	Nb-95	765.8	Fission	3232.	8.36E-03	0.48
r	Cs-134	795.8	Fission	89731.	2.47E-01	14.14
s	Co-58	810.8	Activation	14874.	4.22E-02	2.42
t	Mn-54	834.8	Activation	5051.	1.50E-02	0.86
x	Cs-134	1038.5	Fission	864.	3.65E-03	0.21
y	Fe-59	1099.3	Activation	182.	8.41E-04	0.05
aa	Co-60	1173.2	Activation	11329.	5.81E-02	3.33
bb	Fe-59	1291.6	Activation	156.	9.33E-04	0.05
cc	Co-60	1332.5	Activation	9250.	5.82E-02	3.33
dd	Cs-134	1365.1	Fission	2149.	1.41E-02	0.80
ff	K-40	1460.8	Other	78.	5.70E-04	0.03
ii	Ba-140	1596.5	Fission	154.	1.30E-03	0.07
jj	Co-58	1674.7	Activation	56.	5.13E-04	0.03
kk	Sb-124	1691.0	Activation	45.	4.19E-04	0.02
ll	Rn-222	1764.5	Other	22.	2.20E-04	0.01
mm	Co-56	1771.5	Activation	132.	1.33E-03	0.08
oo	Tl-208	2614.6	Other	19.	3.91E-04	0.02

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
1.79E-01	1.14E+00	4.24E-01	1.75E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !      !       !       !       !       !
ACTIVATION PRODUCTS:     10.25  ■■■■
FISSION PRODUCTS:        65.47  ■■■■■■■■■■■■■■■■■■■■■■■■■■
OTHER PRODUCTION MEANS:  24.28  ■■■■■■■■■■
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	1.16E-01	6.66	64.99
Co-58	Activation	4.27E-02	2.45	23.87
Mn-54	Activation	1.50E-02	0.86	8.40
Fe-59	Activation	1.77E-03	0.10	0.99
Sn-113	Activation	1.39E-03	0.08	0.78
Co-56	Activation	1.33E-03	0.08	0.74
Sb-124	Activation	4.19E-04	0.02	0.23
Cs-134	Fission	6.06E-01	34.67	52.96
Cs-137	Fission	5.22E-01	29.86	45.61
Nb-95	Fission	8.36E-03	0.48	0.73
Zr-95	Fission	3.67E-03	0.21	0.32
I-131	Fission	2.40E-03	0.14	0.21
Ba-140	Fission	1.30E-03	0.07	0.11
Cs-138	Fission	6.33E-04	0.04	0.06
UNKNOWN	Other	4.14E-01	23.69	97.59
ANNIH	Other	9.03E-03	0.52	2.13
K-40	Other	5.70E-04	0.03	0.13
Tl-208	Other	3.91E-04	0.02	0.09
Rn-222	Other	2.20E-04	0.01	0.05

****

****

PLANT: UNIT B RADWASTE BUILDING

LOCATION: PURIFYING ION-EXCHANGER VALVE ROOM/INLET

FILE NAME: POM01 GEOMETRIC CORRECTION FACTOR: 5.61E+03

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 3.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 3.00E-01

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	34865.	1.01E-01	32.33
e	Sn-113	391.7	Activation	837.	6.30E-04	0.20
j	ANNIH	511.0	Other	1256.	1.52E-03	0.49
k	Cs-134	563.3	Fission	906.	1.30E-03	0.42
l	Cs-134	569.3	Fission	1508.	2.20E-03	0.71
m	Cs-134	604.7	Fission	9571.	1.54E-02	4.96
n	Cs-137	661.6	Fission	16896.	3.17E-02	10.18
q	Nb-95	765.8	Fission	529.	1.26E-03	0.41
r	Cs-134	795.8	Fission	9563.	2.43E-02	7.81
s	Co-58	810.8	Activation	1951.	5.11E-03	1.64
t	Mn-54	834.8	Activation	3238.	8.89E-03	2.86
aa	Co-60	1173.2	Activation	11105.	5.26E-02	16.89
bb	Fe-59	1291.6	Activation	15.	8.28E-05	0.03
cc	Co-60	1332.5	Activation	10585.	6.15E-02	19.75
dd	Cs-134	1365.1	Fission	185.	1.12E-03	0.36
ff	K-40	1460.8	Other	306.	2.06E-03	0.66
ll	Rn-222	1764.5	Other	34.	3.14E-04	0.10
oo	Tl-208	2614.6	Other	36.	6.83E-04	0.22

****

****

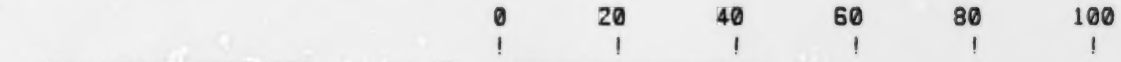
*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
1.29E-01	7.73E-02	1.05E-01	3.11E-01

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***



****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	1.14E-01	36.64	88.57
Mn-54	Activation	8.89E-03	2.86	6.91
Co-58	Activation	5.11E-03	1.64	3.97
Sn-113	Activation	6.30E-04	0.20	0.49
Fe-59	Activation	8.28E-05	0.03	0.06
Cs-134	Fission	4.44E-02	14.25	57.37
Cs-137	Fission	3.17E-02	10.18	41.00
Nb-95	Fission	1.26E-03	0.41	1.63
UNKNOWN	Other	1.01E-01	32.33	95.65
K-40	Other	2.06E-03	0.66	1.96
ANNIH	Other	1.52E-03	0.49	1.45
Tl-208	Other	6.83E-04	0.22	0.65
Rn-222	Other	3.14E-04	0.10	0.30

****

****

PLANT: UNIT B RADWASTE BUILDING

LOCATION: BYPASS ION-EXCHANGER VALVE ROOM/OUTLET

FILE NAME: POM04 GEOMETRIC CORRECTION FACTOR: 4.87E+03

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 24.7

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 2.00E+00

****

****

*** DATA ANALYSIS ***

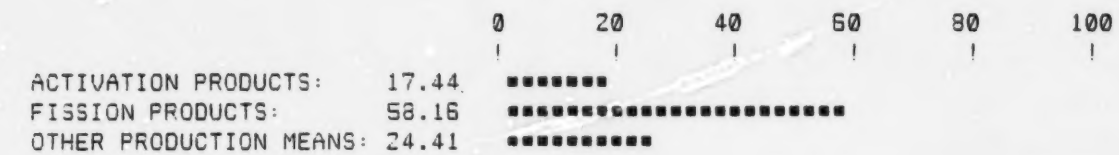
PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	144893.	4.82E-01	23.45
c	Cr-51	320.1	Activation	1392.	8.23E-04	0.04
e	Sn-113	391.7	Activation	6143.	5.32E-03	0.26
h	Cs-134	475.4	Fission	1410.	1.71E-03	0.08
i	Ru-103	497.1	Fission	2066.	2.68E-03	0.13
j	ANNIH	511.0	Other	12317.	1.72E-02	0.84
k	Cs-134	563.3	Fission	15905.	2.62E-02	1.28
l	Cs-134	569.3	Fission	21824.	3.66E-02	1.78
m	Cs-134	604.7	Fission	147709.	2.75E-01	13.37
n	Cs-137	661.6	Fission	237632.	5.14E-01	25.00
o	Zr-95	724.2	Fission	2979.	7.48E-03	0.36
p	Zr-95	756.7	Fission	4612.	1.24E-02	0.61
q	Nb-95	765.8	Fission	12206.	3.36E-02	1.63
r	Cs-134	795.8	Fission	90790.	2.66E-01	12.94
s	Co-58	810.8	Activation	27159.	8.19E-02	3.99
t	Mn-54	834.8	Activation	8971.	2.84E-02	1.38
x	Cs-134	1038.5	Fission	995.	4.47E-03	0.22
y	Fe-59	1099.3	Activation	407.	2.00E-03	0.10
z	Zn-65	1115.5	Activation	287.	1.44E-03	0.07
aa	Co-60	1173.2	Activation	21001.	1.15E-01	5.58
bb	Fe-59	1291.6	Activation	377.	2.40E-03	0.12
cc	Co-60	1332.5	Activation	18008.	1.20E-01	5.86
dd	Cs-134	1365.1	Fission	2209.	1.54E-02	0.75
ff	K-40	1460.8	Other	197.	1.53E-03	0.07
jj	Co-58	1674.7	Activation	51.	4.96E-04	0.02
kk	Sb-124	1691.0	Activation	30.	2.97E-04	0.01
ll	Rn-222	1764.5	Other	19.	2.02E-04	0.01
oo	Tl-208	2614.6	Other	30.	6.56E-04	0.03

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.58E-01	1.19E+00	5.01E-01	2.05E+00

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***



*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.35E-01	11.44	65.62
Co-58	Activation	8.24E-02	4.01	23.02
Mn-54	Activation	2.84E-02	1.38	7.92
Sn-113	Activation	5.32E-03	0.26	1.49
Fe-59	Activation	4.40E-03	0.21	1.23
Zn-65	Activation	1.44E-03	0.07	0.40
Cr-51	Activation	8.23E-04	0.04	0.23
Sb-124	Activation	2.97E-04	0.01	0.08
Cs-134	Fission	6.25E-01	30.42	52.31
Cs-137	Fission	5.14E-01	25.00	42.99
Nb-95	Fission	3.36E-02	1.63	2.81
Zr-95	Fission	1.99E-02	0.97	1.67
Ru-103	Fission	2.68E-03	0.13	0.22
UNKNOWN	Other	4.82E-01	23.45	96.09
ANNIH	Other	1.72E-02	0.84	3.43
K-40	Other	1.53E-03	0.07	0.31
Tl-208	Other	6.56E-04	0.03	0.13
Rn-222	Other	2.02E-04	0.01	0.04

****

PLANT: UNIT B RADWASTE BUILDING

LOCATION: PRESSURIZER OFF-GAS LINE

FILE NAME: POM02 GEOMETRIC CORRECTION FACTOR: 1.18E+04

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 4.3

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 2.00E-01

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	46277.	6.36E-02	30.73
b	Xe-133	81.0	Fission	3800.	2.37E-04	0.11
c	Cr-51	320.1	Activation	76.	1.86E-05	0.01
e	Sn-113	391.7	Activation	292.	1.05E-04	0.05
f	Sb-125	427.9	Fission	325.	1.36E-04	0.07
g	Cs-138	462.8	Fission	116.	5.56E-05	0.03
i	Ru-103	497.1	Fission	335.	1.80E-04	0.09
j	ANNIH	511.0	Other	4454.	2.57E-03	1.24
k	Cs-134	563.3	Fission	1721.	1.17E-03	0.57
l	Cs-134	569.3	Fission	2384.	1.66E-03	0.80
m	Cs-134	604.7	Fission	14422.	1.11E-02	5.36
n	Cs-137	661.6	Fission	24682.	2.21E-02	10.65
o	Zr-95	724.2	Fission	1547.	1.61E-03	0.78
p	Zr-95	756.7	Fission	2517.	2.81E-03	1.36
q	Nb-95	765.8	Fission	6731.	7.66E-03	3.70
r	Cs-134	795.8	Fission	9597.	1.16E-02	5.61
s	Co-58	810.8	Activation	12456.	1.55E-02	7.51
t	Mn-54	834.8	Activation	3781.	4.95E-03	2.39
y	Fe-59	1099.3	Activation	302.	6.14E-04	0.30
aa	Co-60	1173.2	Activation	11980.	2.70E-02	13.05
bb	Fe-59	1291.6	Activation	188.	4.95E-04	0.24
cc	Co-60	1332.5	Activation	10707.	2.96E-02	14.30
dd	Cs-134	1365.1	Fission	226.	6.50E-04	0.31
ff	K-40	1460.8	Other	284.	9.12E-04	0.44
jj	Co-58	1674.7	Activation	19.	7.65E-05	0.04
kk	Sb-124	1691.0	Activation	51.	2.09E-04	0.10
ll	Rn-222	1764.5	Other	30.	1.32E-04	0.06
nn	Rn-222	2204.1	Other	4.	2.61E-05	0.01
oo	Tl-208	2614.6	Other	27.	2.44E-04	0.12

****

****

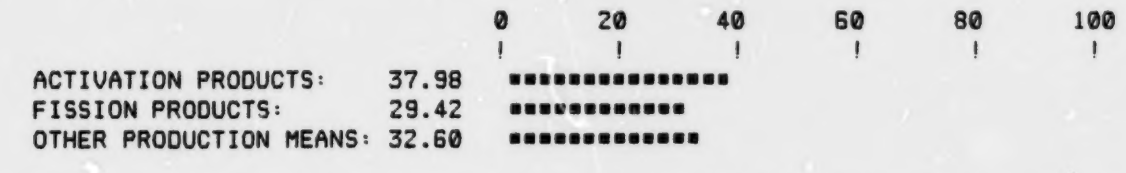
*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
7.87E-02	6.09E-02	6.75E-02	2.07E-01

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***



****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	5.67E-02	27.35	72.02
Co-58	Activation	1.56E-02	7.54	19.86
Mn-54	Activation	4.95E-03	2.39	6.29
Fe-59	Activation	1.11E-03	0.54	1.41
Sb-124	Activation	2.09E-04	0.10	0.27
Sn-113	Activation	1.05E-04	0.05	0.13
Cr-51	Activation	1.86E-05	0.01	0.02
Cs-134	Fission	2.62E-02	12.65	42.98
Cs-137	Fission	2.21E-02	10.65	36.21
Nb-95	Fission	7.66E-03	3.70	12.56
Zr-95	Fission	4.41E-03	2.13	7.24
Xe-133	Fission	2.37E-04	0.11	0.39
Ru-103	Fission	1.80E-04	0.09	0.30
Sb-125	Fission	1.36E-04	0.07	0.22
Cs-138	Fission	5.56E-05	0.03	0.09
UNKNOWN	Other	6.36E-02	30.73	94.25
ANNIH	Other	2.57E-03	1.24	3.81
K-40	Other	9.12E-04	0.44	1.35
Tl-208	Other	2.44E-04	0.12	0.36
Rn-222	Other	1.58E-04	0.08	0.23

****

****

PLANT: UNIT B RADWASTE BUILDING

LOCATION: PRESSURIZER LEVEL CONTROL VALVE ROOM

FILE NAME: POM03 GEOMETRIC CORRECTION FACTOR: 4.20E+03

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 16.7

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 1.50E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	138781.	5.34E-01	34.25
c	Cr-51	320.1	Activation	491.	3.36E-04	0.02
e	Sn-113	391.7	Activation	7379.	7.40E-03	0.47
f	Sb-125	427.9	Fission	336.	3.95E-04	0.03
h	Cs-134	475.4	Fission	1.	1.40E-06	0.00
i	Ru-103	497.1	Fission	1520.	2.29E-03	0.15
j	ANNIH	511.0	Other	13422.	2.17E-02	1.39
k	Cs-134	563.3	Fission	5300.	1.01E-02	0.65
l	Cs-134	569.3	Fission	8947.	1.74E-02	1.11
m	Cs-134	604.7	Fission	54929.	1.18E-01	7.58
n	Cs-137	661.6	Fission	94276.	2.36E-01	15.12
o	Zr-95	724.2	Fission	4636.	1.35E-02	0.86
p	Zr-95	756.7	Fission	6221.	1.94E-02	1.25
q	Nb-95	765.8	Fission	20056.	6.38E-02	4.09
r	Cs-134	795.8	Fission	33399.	1.13E-01	7.26
s	Co-58	810.8	Activation	29828.	1.04E-01	6.68
t	Mn-54	834.8	Activation	5306.	1.94E-02	1.25
x	Cs-134	1038.5	Fission	477.	2.48E-03	0.16
y	Fe-59	1099.3	Activation	726.	4.13E-03	0.26
z	Zn-65	1115.5	Activation	2657.	1.55E-02	0.99
aa	Co-60	1173.2	Activation	18243.	1.15E-01	7.38
bb	Fe-59	1291.6	Activation	293.	2.16E-03	0.14
cc	Co-60	1332.5	Activation	16420.	1.27E-01	8.15
dd	Cs-134	1365.1	Fission	614.	4.94E-03	0.32
ff	K-40	1460.8	Other	349.	3.14E-03	0.20
jj	Co-58	1674.7	Activation	43.	4.84E-04	0.03
kk	Sb-124	1691.0	Activation	124.	1.42E-03	0.09
ll	Rn-222	1764.5	Other	28.	3.45E-04	0.02
nn	Rn-222	2204.1	Other	31.	5.65E-04	0.04
oo	Tl-208	2614.6	Other	35.	8.86E-04	0.06

****

****

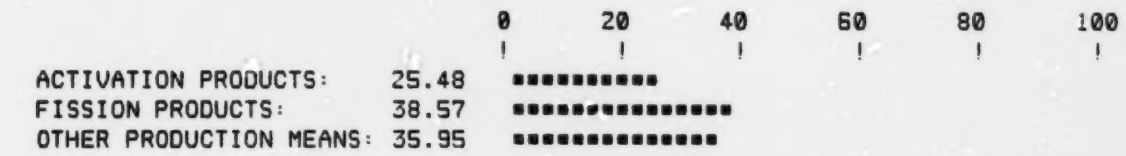
*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.97E-01	6.02E-01	5.61E-01	1.56E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***



****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.42E-01	15.54	60.99
Co-58	Activation	1.05E-01	6.71	26.34
Mn-54	Activation	1.94E-02	1.25	4.89
Zn-65	Activation	1.55E-02	0.99	3.90
Sn-113	Activation	7.40E-03	0.47	1.86
Fe-59	Activation	6.29E-03	0.40	1.58
Sb-124	Activation	1.42E-03	0.09	0.36
Cr-51	Activation	3.36E-04	0.02	0.08
Cs-134	Fission	2.66E-01	17.08	44.27
Cs-137	Fission	2.36E-01	15.12	39.21
Nb-95	Fission	6.38E-02	4.09	10.61
Zr-95	Fission	3.29E-02	2.11	5.47
Ru-103	Fission	2.29E-03	0.15	0.38
Sb-125	Fission	3.95E-04	0.03	0.07
UNKNOWN	Other	5.34E-01	34.25	95.25
ANNIH	Other	2.17E-02	1.39	3.87
K-40	Other	3.14E-03	0.20	0.56
Rn-222	Other	9.10E-04	0.06	0.16
Tl-208	Other	8.86E-04	0.06	0.16

****

****

PLANT: UNIT B RADWASTE BUILDING

LOCATION: CHARGING PUMP ROOM

FILE NAME: POM05 GEOMETRIC CORRECTION FACTOR: 7.98E+03

LIVE COUNTING TIME (SEC): 600. DEAD TIME (%): 1.2

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 1.00E-01

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	34834.	3.53E-02	33.98
e	Sn-113	391.7	Activation	891.	2.36E-04	0.23
h	Cs-134	475.4	Fission	224.	8.27E-05	0.08
j	ANNIH	511.0	Other	1309.	5.57E-04	0.54
k	Cs-134	563.3	Fission	1206.	6.07E-04	0.58
l	Cs-134	569.3	Fission	2008.	1.03E-03	0.99
m	Cs-134	604.7	Fission	12806.	7.26E-03	6.99
n	Cs-137	661.6	Fission	40799.	2.69E-02	25.87
o	Zr-95	724.2	Fission	247.	1.89E-04	0.18
p	Zr-95	756.7	Fission	1823.	1.50E-03	1.44
q	Nb-95	765.8	Fission	3151.	2.64E-03	2.54
r	Cs-134	795.8	Fission	8041.	7.18E-03	6.91
s	Co-58	810.8	Activation	1502.	1.38E-03	1.33
t	Mn-54	834.8	Activation	1151.	1.11E-03	1.07
x	Cs-134	1038.5	Fission	107.	1.47E-04	0.14
y	Fe-59	1099.3	Activation	97.	1.45E-04	0.14
aa	Co-60	1173.2	Activation	4084.	6.79E-03	6.54
bb	Fe-59	1291.6	Activation	177.	3.43E-04	0.33
cc	Co-60	1332.5	Activation	3866.	7.89E-03	7.59
dd	Cs-134	1365.1	Fission	202.	4.28E-04	0.41
ff	K-40	1460.8	Other	598.	1.42E-03	1.36
hh	Th-232	1587.9	Other	52.	1.41E-04	0.14
ll	Rn-222	1764.5	Other	43.	1.39E-04	0.13
oo	Tl-208	2614.6	Other	79.	5.27E-04	0.51

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
1.79E-02	4.80E-02	3.81E-02	1.04E-01

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !       !        !        !        !        !
ACTIVATION PRODUCTS:     17.22  *******
FISSION PRODUCTS:        46.13  ******************
OTHER PRODUCTION MEANS:  36.65  **************
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	1.47E-02	14.12	82.03
Co-58	Activation	1.38E-03	1.33	7.72
Mn-54	Activation	1.11E-03	1.07	6.20
Fe-59	Activation	4.89E-04	0.47	2.73
Sn-113	Activation	2.36E-04	0.23	1.32
Cs-137	Fission	2.69E-02	25.87	56.08
Cs-134	Fission	1.67E-02	16.10	34.89
Nb-95	Fission	2.64E-03	2.54	5.51
Zr-95	Fission	1.69E-03	1.62	3.52
UNKNOWN	Other	3.53E-02	33.98	92.70
K-40	Other	1.42E-03	1.36	3.72
ANNIH	Other	5.57E-04	0.54	1.46
Tl-208	Other	5.27E-04	0.51	1.38
Th-232	Other	1.41E-04	0.14	0.37
Rn-222	Other	1.39E-04	0.13	0.37

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: PRESSURIZER SURGE LINE

FILE NAME: POM16 GEOMETRIC CORRECTION FACTOR: 1.67E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 93.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 8.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	27502.	7.99E-01	9.87
i	Ru-103	497.1	Fission	168.	1.91E-03	0.02
j	ANNIH	511.0	Other	9878.	1.20E-01	1.49
k	Cs-134	563.3	Fission	3355.	4.83E-02	0.60
l	Cs-134	569.3	Fission	6151.	9.02E-02	1.12
m	Cs-134	604.7	Fission	42342.	6.88E-01	8.50
n	Cs-137	661.6	Fission	71295.	1.35E+00	16.64
o	Zr-95	724.2	Fission	4496.	9.86E-02	1.22
p	Zr-95	756.7	Fission	3080.	7.25E-02	0.90
q	Nb-95	765.8	Fission	12840.	3.08E-01	3.81
r	Cs-134	795.8	Fission	28551.	7.30E-01	9.02
s	Co-58	810.8	Activation	23781.	6.27E-01	7.75
t	Mn-54	834.8	Activation	11397.	3.15E-01	3.89
x	Cs-134	1038.5	Fission	1.	3.92E-05	0.00
y	Fe-59	1099.3	Activation	44.	1.89E-03	0.02
aa	Co-60	1173.2	Activation	26738.	1.27E+00	15.75
bb	Fe-59	1291.6	Activation	439.	2.44E-02	0.30
cc	Co-60	1332.5	Activation	25428.	1.49E+00	18.37
dd	Cs-134	1365.1	Fission	823.	5.00E-02	0.62
jj	Co-58	1674.7	Activation	105.	8.93E-03	0.11

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.74E+00	3.43E+00	9.19E-01	8.09E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !        !        !        !        !        !
ACTIVATION PRODUCTS:    46.19   ******************
FISSION PRODUCTS:       42.45   ******************
OTHER PRODUCTION MEANS: 11.36   *****
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.76E+00	34.11	73.86
Co-58	Activation	6.36E-01	7.86	17.01
Mn-54	Activation	3.15E-01	3.89	8.43
Fe-59	Activation	2.63E-02	0.32	0.70
Cs-134	Fission	1.61E+00	19.86	46.78
Cs-137	Fission	1.35E+00	16.64	39.20
Nb-95	Fission	3.08E-01	3.81	8.98
Zr-95	Fission	1.71E-01	2.12	4.98
Ru-103	Fission	1.91E-03	0.02	0.06
UNKNOWN	Other	7.99E-01	9.87	86.90
ANNIH	Other	1.20E-01	1.49	13.10

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: HOT LEG OF STEAM GENERATOR #2

FILE NAME: POM13 GEOMETRIC CORRECTION FACTOR: 1.77E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 63.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 5.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	4757.	1.31E-01	2.60
j	ANNIH	511.0	Other	4407.	5.08E-02	1.01
k	Cs-134	563.3	Fission	2902.	3.95E-02	0.79
l	Cs-134	569.3	Fission	4363.	6.05E-02	1.21
m	Cs-134	604.7	Fission	27157.	4.17E-01	8.31
n	Cs-137	661.6	Fission	52243.	9.32E-01	18.59
o	Zr-95	724.2	Fission	277.	5.74E-03	0.11
p	Zr-95	756.7	Fission	328.	7.30E-03	0.15
q	Nb-95	765.8	Fission	2339.	5.31E-02	1.06
r	Cs-134	795.8	Fission	18055.	4.36E-01	8.70
s	Co-58	810.8	Activation	23250.	5.79E-01	11.55
t	Mn-54	834.8	Activation	4469.	1.17E-01	2.33
y	Fe-59	1099.3	Activation	342.	1.39E-02	0.28
z	Zn-65	1115.5	Activation	211.	8.77E-03	0.17
aa	Co-60	1173.2	Activation	21073.	9.49E-01	18.92
bb	Fe-59	1291.6	Activation	350.	1.84E-02	0.37
cc	Co-60	1332.5	Activation	20809.	1.15E+00	22.92
dd	Cs-134	1365.1	Fission	585.	3.36E-02	0.67
jj	Co-58	1674.7	Activation	166.	1.33E-02	0.27

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
2.85E+00	1.99E+00	1.81E-01	5.01E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                  0       20       40       60       80       100
                                  !        !        !        !        !        !
ACTIVATION PRODUCTS:     56.80    ************************
FISSION PRODUCTS:        39.58    ****************
OTHER PRODUCTION MEANS:   3.62    *
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.10E+00	41.84	73.66
Co-58	Activation	5.93E-01	11.82	20.80
Mn-54	Activation	1.17E-01	2.33	4.10
Fe-59	Activation	3.23E-02	0.64	1.13
Zn-65	Activation	8.77E-03	0.17	0.31
Cs-134	Fission	9.87E-01	19.68	49.71
Cs-137	Fission	9.32E-01	18.59	46.96
Nb-95	Fission	5.31E-02	1.06	2.67
Zr-95	Fission	1.30E-02	0.26	0.66
UNKNOWN	Other	1.31E-01	2.60	72.00
ANNIH	Other	5.08E-02	1.01	28.00

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: VALVE NEAR STEAM GENERATOR #2

FILE NAME: POM08 GEOMETRIC CORRECTION FACTOR: 2.80E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 63.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 6.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	10267.	1.78E-01	2.96
g	Cs-138	462.8	Fission	190.	1.15E-03	0.02
j	ANNIH	511.0	Other	3302.	2.40E-02	0.40
k	Cs-134	563.3	Fission	5736.	4.93E-02	0.82
l	Cs-134	569.3	Fission	10953.	9.59E-02	1.59
m	Cs-134	604.7	Fission	63861.	6.19E-01	10.28
n	Cs-137	661.6	Fission	113022.	1.27E+00	21.15
o	Zr-95	724.2	Fission	1305.	1.71E-02	0.28
p	Zr-95	756.7	Fission	1022.	1.44E-02	0.24
q	Nb-95	765.8	Fission	4565.	6.54E-02	1.09
r	Cs-134	795.8	Fission	40393.	6.16E-01	10.24
s	Co-58	810.8	Activation	8842.	1.39E-01	2.31
t	Mn-54	834.8	Activation	10347.	1.71E-01	2.83
y	Fe-59	1099.3	Activation	193.	4.95E-03	0.08
aa	Co-60	1173.2	Activation	43954.	1.25E+00	20.76
bb	Fe-59	1291.6	Activation	199.	6.60E-03	0.11
cc	Co-60	1332.5	Activation	41807.	1.46E+00	24.22
dd	Cs-134	1365.1	Fission	1034.	3.75E-02	0.62

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.03E+00	2.79E+00	2.02E-01	6.02E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !        !        !        !        !        !
ACTIVATION PRODUCTS:    50.31   ■■■■■■■■■■■■■■■■■■■■
FISSION PRODUCTS:       46.33   ■■■■■■■■■■■■■■■■■■
OTHER PRODUCTION MEANS:  3.35   ■
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.71E+00	44.97	89.39
Mn-54	Activation	1.71E-01	2.83	5.63
Co-58	Activation	1.39E-01	2.31	4.59
Fe-59	Activation	1.15E-02	0.19	0.38
Cs-134	Fission	1.42E+00	23.55	50.84
Cs-137	Fission	1.27E+00	21.15	45.65
Nb-95	Fission	6.54E-02	1.09	2.35
Zr-95	Fission	3.14E-02	0.52	1.13
Cs-138	Fission	1.15E-03	0.02	0.04
UNKNOWN	Other	1.78E-01	2.96	88.11
ANNIH	Other	2.40E-02	0.40	11.89

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: LETDOWN VALVE NEAR STEAM GENERATOR #1

FILE NAME: POM10 GEOMETRIC CORRECTION FACTOR: 2.46E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 69.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 5.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	64675.	1.27E+00	24.78
d	I-131	364.5	Fission	3.	1.35E-05	0.00
e	Sn-113	391.7	Activation	1606.	8.25E-03	0.16
i	Ru-103	497.1	Fission	904.	6.96E-03	0.14
j	ANNIH	511.0	Other	14173.	1.17E-01	2.28
k	Cs-134	563.3	Fission	578.	5.65E-03	0.11
l	Cs-134	569.3	Fission	667.	6.64E-03	0.13
m	Cs-134	604.7	Fission	4524.	4.98E-02	0.97
n	Cs-137	661.6	Fission	8343.	1.07E-01	2.08
o	Zr-95	724.2	Fission	4383.	6.52E-02	1.27
p	Zr-95	756.7	Fission	5023.	8.03E-02	1.56
q	Nb-95	765.8	Fission	17386.	2.83E-01	5.51
r	Cs-134	795.8	Fission	2398.	4.16E-02	0.81
s	Co-58	810.8	Activation	38504.	6.89E-01	13.39
t	Mn-54	834.8	Activation	9884.	1.85E-01	3.60
y	Fe-59	1099.3	Activation	574.	1.67E-02	0.33
aa	Co-60	1173.2	Activation	30683.	9.92E-01	19.29
bb	Fe-59	1291.6	Activation	463.	1.75E-02	0.34
cc	Co-60	1332.5	Activation	29878.	1.18E+00	23.03
jj	Co-58	1674.7	Activation	215.	1.24E-02	0.24

****

BEST COPY AVAILABLE

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.10E+00	6.46E-01	1.39E+00	5.14E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !      !       !       !       !       !
ACTIVATION PRODUCTS:    60.37   ************************
FISSION PRODUCTS:       12.57   *****
OTHER PRODUCTION MEANS: 27.06   ***********
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.18E+00	42.32	70.09
Co-58	Activation	7.01E-01	13.63	22.58
Mn-54	Activation	1.85E-01	3.60	5.97
Fe-59	Activation	3.42E-02	0.66	1.10
Sn-113	Activation	8.25E-03	0.16	0.27
Nb-95	Fission	2.83E-01	5.51	43.83
Zr-95	Fission	1.45E-01	2.83	22.50
Cs-137	Fission	1.07E-01	2.08	16.53
Cs-134	Fission	1.04E-01	2.02	16.05
Ru-103	Fission	6.96E-03	0.14	1.08
I-131	Fission	1.35E-05	0.00	0.00
UNKNOWN	Other	1.27E+00	24.78	91.58
ANNIH	Other	1.17E-01	2.28	8.42

****

BEST COPY AVAILABLE

****

PLANT: UNIT B CONTAINMENT

LOCATION: SHUTDOWN COOLING LINE - 17' 6'' LEVEL

FILE NAME: POM09 GEOMETRIC CORRECTION FACTOR: 1.87E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 82.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 6.50E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	5587.	1.45E-01	2.23
c	Cr-51	320.1	Activation	1308.	6.05E-03	0.09
e	Sn-113	391.7	Activation	2138.	1.45E-02	0.22
i	Ru-103	497.1	Fission	540.	5.49E-03	0.08
j	ANNIH	511.0	Other	9212.	1.01E-01	1.54
k	Cs-134	563.3	Fission	5326.	6.87E-02	1.05
l	Cs-134	569.3	Fission	8345.	1.10E-01	1.68
m	Cs-134	604.7	Fission	55233.	8.03E-01	12.33
n	Cs-137	661.6	Fission	77193.	1.31E+00	20.03
o	Zr-95	724.2	Fission	4468.	8.77E-02	1.35
p	Zr-95	756.7	Fission	3940.	8.31E-02	1.28
q	Nb-95	765.8	Fission	16496.	3.55E-01	5.45
r	Cs-134	795.8	Fission	36756.	8.42E-01	12.92
s	Co-58	810.8	Activation	35412.	8.36E-01	12.83
t	Mn-54	834.8	Activation	5765.	1.43E-01	2.19
x	Cs-134	1038.5	Fission	238.	8.36E-03	0.13
y	Fe-59	1099.3	Activation	876.	3.37E-02	0.52
aa	Co-60	1173.2	Activation	16980.	7.25E-01	11.12
bb	Fe-59	1291.6	Activation	653.	3.25E-02	0.50
cc	Co-60	1332.5	Activation	13826.	7.24E-01	11.10
dd	Cs-134	1365.1	Fission	1181.	6.43E-02	0.99
jj	Co-58	1674.7	Activation	182.	1.39E-02	0.21
kk	Sb-124	1691.0	Activation	141.	1.09E-02	0.17

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
2.54E+00	3.73E+00	2.46E-01	6.52E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !       !       !       !       !       !
ACTIVATION PRODUCTS:    38.95   ***************
FISSION PRODUCTS:       57.28   ***********************
OTHER PRODUCTION MEANS:  3.77   **
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	1.45E+00	22.22	57.05
Co-58	Activation	8.50E-01	13.04	33.48
Mn-54	Activation	1.43E-01	2.19	5.62
Fe-59	Activation	6.62E-02	1.02	2.61
Sn-113	Activation	1.45E-02	0.22	0.57
Sb-124	Activation	1.09E-02	0.17	0.43
Cr-51	Activation	6.05E-03	0.09	0.24
•				
Cs-134	Fission	1.90E+00	29.10	50.80
Cs-137	Fission	1.31E+00	20.03	34.97
Nb-95	Fission	3.55E-01	5.45	9.51
Zr-95	Fission	1.71E-01	2.62	4.58
Ru-103	Fission	5.49E-03	0.08	0.15
UNKNOWN	Other	1.45E-01	2.23	59.10
ANNIH	Other	1.01E-01	1.54	40.90

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: SHUTDOWN COOLING LINE - 30' LEVEL

FILE NAME: POM12 GEOMETRIC CORRECTION FACTOR: 2.74E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 41.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 4.10E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	2094.	3.71E-02	0.90
e	Sn-113	391.7	Activation	203.	9.37E-04	0.02
h	Cs-134	475.4	Fission	1037.	6.68E-03	0.16
j	ANNIH	511.0	Other	2326.	1.73E-02	0.42
k	Cs-134	563.3	Fission	6685.	5.87E-02	1.43
l	Cs-134	569.3	Fission	10112.	9.04E-02	2.20
m	Cs-134	604.7	Fission	67675.	6.70E-01	16.33
n	Cs-137	661.6	Fission	128644.	1.48E+00	36.09
o	Zr-95	724.2	Fission	766.	1.02E-02	0.25
p	Zr-95	756.7	Fission	487.	7.00E-03	0.17
q	Nb-95	765.8	Fission	2490.	3.65E-02	0.89
r	Cs-134	795.8	Fission	45569.	7.11E-01	17.31
s	Co-58	810.8	Activation	7933.	1.28E-01	3.11
t	Mn-54	834.8	Activation	3594.	6.06E-02	1.48
x	Cs-134	1038.5	Fission	808.	1.93E-02	0.47
y	Fe-59	1099.3	Activation	443.	1.16E-02	0.28
aa	Co-60	1173.2	Activation	11587.	3.37E-01	8.20
bb	Fe-59	1291.6	Activation	171.	5.79E-03	0.14
cc	Co-60	1332.5	Activation	10312.	3.67E-01	8.95
dd	Cs-134	1365.1	Fission	1239.	4.59E-02	1.12
jj	Co-58	1674.7	Activation	44.	2.28E-03	0.06
kk	Sb-124	1691.0	Activation	7.	3.69E-04	0.01

****

****

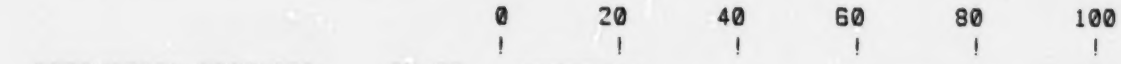
*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
9.13E-01	3.14E+00	5.44E-02	4.10E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***



****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	7.04E-01	17.16	77.11
Co-58	Activation	1.30E-01	3.16	14.21
Mn-54	Activation	6.06E-02	1.48	6.63
Fe-59	Activation	1.74E-02	0.42	1.91
Sn-113	Activation	9.37E-04	0.02	0.10
Sb-124	Activation	3.69E-04	0.01	0.04
Cs-134	Fission	1.60E+00	39.03	51.07
Cs-137	Fission	1.48E+00	36.09	47.22
Nb-95	Fission	3.65E-02	0.89	1.16
Zr-95	Fission	1.72E-02	0.42	0.55
UNKNOWN	Other	3.71E-02	0.90	68.20
ANNIH	Other	1.73E-02	0.42	31.80

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: SHUTDOWN COOLING LINE UPSTREAM OF POM14 VALVE

FILE NAME: POM15 GEOMETRIC CORRECTION FACTOR: 1.67E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 47.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 4.20E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	14647.	4.25E-01	10.00
i	Ru-103	497.1	Fission	112.	1.27E-03	0.03
j	ANNIH	511.0	Other	4308.	5.25E-02	1.24
k	Cs-134	563.3	Fission	2077.	2.99E-02	0.70
l	Cs-134	569.3	Fission	3082.	4.52E-02	1.06
m	Cs-134	604.7	Fission	21784.	3.53E-01	8.32
n	Cs-137	661.6	Fission	40021.	7.55E-01	17.77
o	Zr-95	724.2	Fission	35.	7.66E-04	0.02
p	Zr-95	756.7	Fission	203.	4.78E-03	0.11
q	Nb-95	765.8	Fission	2400.	5.76E-02	1.36
r	Cs-134	795.8	Fission	16923.	4.32E-01	10.18
s	Co-58	810.8	Activation	18448.	4.86E-01	11.44
t	Mn-54	834.8	Activation	3421.	9.44E-02	2.22
x	Cs-134	1038.5	Fission	91.	3.57E-03	0.08
y	Fe-59	1099.3	Activation	547.	2.35E-02	0.55
aa	Co-60	1173.2	Activation	12940.	6.16E-01	14.50
bb	Fe-59	1291.6	Activation	616.	3.42E-02	0.81
cc	Co-60	1332.5	Activation	13254.	7.74E-01	18.21
dd	Cs-134	1365.1	Fission	746.	4.53E-02	1.07
jj	Co-58	1674.7	Activation	167.	1.42E-02	0.33

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
2.04E+00	1.73E+00	4.77E-01	4.25E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !        !        !        !        !        !
ACTIVATION PRODUCTS:    48.06   ********************
FISSION PRODUCTS:       40.70   ****************
OTHER PRODUCTION MEANS: 11.24   ****
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	1.39E+00	32.71	68.06
Co-58	Activation	5.00E-01	11.77	24.49
Mn-54	Activation	9.44E-02	2.22	4.63
Fe-59	Activation	5.77E-02	1.36	2.82
Cs-134	Fission	9.10E-01	21.41	52.62
Cs-137	Fission	7.55E-01	17.77	43.66
Nb-95	Fission	5.76E-02	1.36	3.33
Zr-95	Fission	5.54E-03	0.13	0.32
Ru-103	Fission	1.27E-03	0.03	0.07
UNKNOWN	Other	4.25E-01	10.00	89.01
ANNIH	Other	5.25E-02	1.24	10.99

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: SHUTDOWN COOLING ISOLATION VALVE

FILE NAME: POM14 GEOMETRIC CORRECTION FACTOR: 2.59E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 75.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 6.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	103021.	1.93E+00	31.04
j	ANNIH	511.0	Other	5882.	4.62E-02	0.74
k	Cs-134	563.3	Fission	2725.	2.53E-02	0.41
l	Cs-134	569.3	Fission	4884.	4.62E-02	0.74
m	Cs-134	604.7	Fission	32620.	3.42E-01	5.50
n	Cs-137	661.6	Fission	99383.	1.21E+00	19.46
p	Zr-95	756.7	Fission	639.	9.71E-03	0.16
q	Nb-95	765.8	Fission	2248.	3.48E-02	0.56
r	Cs-134	795.8	Fission	20075.	3.31E-01	5.33
s	Co-58	810.8	Activation	19638.	3.34E-01	5.37
t	Mn-54	834.8	Activation	4572.	8.15E-02	1.31
aa	Co-60	1173.2	Activation	25412.	7.81E-01	12.56
bb	Fe-59	1291.6	Activation	129.	4.62E-03	0.07
cc	Co-60	1332.5	Activation	26786.	1.01E+00	16.24
dd	Cs-134	1365.1	Fission	542.	2.12E-02	0.34
jj	Co-58	1674.7	Activation	191.	1.05E-02	0.17

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
2.22E+00	2.02E+00	1.98E+00	6.22E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                  0       20       40       60       80       100
                                  !       !        !        !        !        !
ACTIVATION PRODUCTS:     35.72    ***************
FISSION PRODUCTS:        32.49    *************
OTHER PRODUCTION MEANS:  31.78    *************
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	1.79E+00	28.80	80.62
Co-58	Activation	3.44E-01	5.54	15.50
Mn-54	Activation	8.15E-02	1.31	3.67
Fe-59	Activation	4.62E-03	0.07	0.21
Cs-137	Fission	1.21E+00	19.46	59.90
Cs-134	Fission	7.65E-01	12.31	37.89
Nb-95	Fission	3.48E-02	0.56	1.72
Zr-95	Fission	9.71E-03	0.16	0.48
UNKNOWN	Other	1.93E+00	31.04	97.66
ANNIH	Other	4.62E-02	0.74	2.34

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: CAVITY DRAIN LINE NEAR ELEVATION ACCESS POINT

FILE NAME: POM11 GEOMETRIC CORRECTION FACTOR: 3.73E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 40.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 4.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	55724.	7.25E-01	17.76
c	Cr-51	320.1	Activation	7803.	1.80E-02	0.44
e	Sn-113	391.7	Activation	748.	2.54E-03	0.06
f	Sb-125	427.9	Fission	480.	1.91E-03	0.05
i	Ru-103	497.1	Fission	226.	1.15E-03	0.03
j	ANNIH	511.0	Other	15149.	8.27E-02	2.03
k	Cs-134	563.3	Fission	2370.	1.53E-02	0.37
l	Cs-134	569.3	Fission	4558.	2.99E-02	0.73
m	Cs-134	604.7	Fission	25883.	1.88E-01	4.61
n	Cs-137	661.6	Fission	29843.	2.52E-01	6.18
o	Zr-95	724.2	Fission	5120.	5.03E-02	1.23
p	Zr-95	756.7	Fission	4318.	4.56E-02	1.12
q	Nb-95	765.8	Fission	17808.	1.92E-01	4.69
r	Cs-134	795.8	Fission	15232.	1.74E-01	4.27
s	Co-58	810.8	Activation	37039.	4.37E-01	10.71
t	Mn-54	834.8	Activation	13675.	1.69E-01	4.15
y	Fe-59	1099.3	Activation	2392.	4.60E-02	1.13
aa	Co-60	1173.2	Activation	34949.	7.46E-01	18.28
bb	Fe-59	1291.6	Activation	1282.	3.19E-02	0.78
cc	Co-60	1332.5	Activation	32510.	8.51E-01	20.85
dd	Cs-134	1365.1	Fission	488.	1.33E-02	0.33
jj	Co-58	1674.7	Activation	161.	6.13E-03	0.15
kk	Sb-124	1691.0	Activation	40.	1.55E-03	0.04

****

101

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
2.31E+00	9.64E-01	8.08E-01	4.08E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !      !       !       !       !       !
ACTIVATION PRODUCTS:    56.59   ■■■■■■■■■■■■■■■■■■■■■■■
FISSION PRODUCTS:       23.62   ■■■■■■■■■
OTHER PRODUCTION MEANS: 19.79   ■■■■■■■■
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	1.60E+00	39.12	69.14
Co-58	Activation	4.43E-01	10.86	19.20
Mn-54	Activation	1.69E-01	4.15	7.33
Fe-59	Activation	7.79E-02	1.91	3.37
Cr-51	Activation	1.80E-02	0.44	0.78
Sn-113	Activation	2.54E-03	0.06	0.11
Sb-124	Activation	1.55E-03	0.04	0.07
Cs-134	Fission	4.21E-01	10.32	43.70
Cs-137	Fission	2.52E-01	6.18	26.18
Nb-95	Fission	1.92E-01	4.69	19.87
Zr-95	Fission	9.58E-02	2.35	9.94
Sb-125	Fission	1.91E-03	0.05	0.20
Ru-103	Fission	1.15E-03	0.03	0.12
UNKNOWN	Other	7.25E-01	17.76	89.76
ANNIH	Other	8.27E-02	2.03	10.24

****

****

PLANT: UNIT B CONTAINMENT

LOCATION: MAIN CAVITY DRAIN LINE

FILE NAME: POM07 GEOMETRIC CORRECTION FACTOR: 3.85E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 47.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 4.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	70822.	8.92E-01	21.76
c	Cr-51	320.1	Activation	17287.	3.87E-02	0.94
i	Ru-103	497.1	Fission	853.	4.20E-03	0.10
j	ANNIH	511.0	Other	12941.	6.84E-02	1.67
m	Cs-134	604.7	Fission	2324.	1.64E-02	0.40
n	Cs-137	661.6	Fission	2103.	1.72E-02	0.42
o	Zr-95	724.2	Fission	443.	4.21E-03	0.10
p	Zr-95	756.7	Fission	2656.	2.71E-02	0.66
q	Nb-95	765.8	Fission	7155.	7.45E-02	1.82
s	Co-58	810.8	Activation	41370.	4.73E-01	11.54
t	Mn-54	834.8	Activation	13243.	1.59E-01	3.87
y	Fe-59	1099.3	Activation	58.	1.08E-03	0.03
aa	Co-60	1173.2	Activation	52617.	1.09E+00	26.52
bb	Fe-59	1291.6	Activation	1356.	3.27E-02	0.80
cc	Co-60	1332.5	Activation	47161.	1.20E+00	29.15
ff	K-40	1460.8	Other	31.	9.12E-04	0.02
ii	Ba-140	1596.5	Fission	136.	4.63E-03	0.11
kk	Sb-124	1691.0	Activation	51.	1.91E-03	0.05
oo	Tl-208	2614.6	Other	19.	1.57E-03	0.04

****

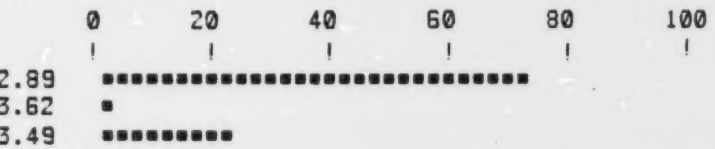
****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
2.99E+00	1.48E-01	9.63E-01	4.10E+00

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***



*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.28E+00	55.67	76.38
Co-58	Activation	4.73E-01	11.54	15.83
Mn-54	Activation	1.59E-01	3.87	5.31
Cr-51	Activation	3.87E-02	0.94	1.29
Fe-59	Activation	3.38E-02	0.82	1.13
Sb-124	Activation	1.91E-03	0.05	0.06
Nb-95	Fission	7.45E-02	1.82	50.26
Zr-95	Fission	3.13E-02	0.76	21.14
Cs-137	Fission	1.72E-02	0.42	11.61
Cs-134	Fission	1.64E-02	0.40	11.04
Ba-140	Fission	4.63E-03	0.11	3.12
Ru-103	Fission	4.20E-03	0.10	2.83
UNKNOWN	Other	8.92E-01	21.76	92.64
ANNIH	Other	6.84E-02	1.67	7.11
Tl-208	Other	1.57E-03	0.04	0.16
K-40	Other	9.12E-04	0.02	0.09

****

PLANT: UNIT B PENETRATION BUILDING

LOCATION: SPENT FUEL POOL PUMP ROOM

FILE NAME: POM17 GEOMETRIC CORRECTION FACTOR: 2.34E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 24.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 3.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	12338.	2.55E-01	8.43
c	Cr-51	320.1	Activation	3392.	1.25E-02	0.41
e	Sn-113	391.7	Activation	1836.	9.91E-03	0.33
i	Ru-103	497.1	Fission	223.	1.80E-03	0.06
j	ANNIH	511.0	Other	16231.	1.41E-01	4.66
k	Cs-134	563.3	Fission	802.	8.23E-03	0.27
l	Cs-134	569.3	Fission	3251.	3.40E-02	1.12
m	Cs-134	604.7	Fission	21296.	2.47E-01	8.14
n	Cs-137	661.6	Fission	23196.	3.12E-01	10.31
o	Zr-95	724.2	Fission	3322.	5.19E-02	1.71
p	Zr-95	756.7	Fission	3381.	5.68E-02	1.87
q	Nb-95	765.8	Fission	10714.	1.83E-01	6.06
r	Cs-134	795.8	Fission	14091.	2.57E-01	8.48
s	Co-58	810.8	Activation	38352.	7.21E-01	23.79
t	Mn-54	834.8	Activation	4008.	7.89E-02	2.61
v	Ag-110m	884.7	Activation	37.	8.00E-04	0.03
y	Fe-59	1099.3	Activation	802.	2.45E-02	0.81
aa	Co-60	1173.2	Activation	8173.	2.78E-01	9.16
bb	Fe-59	1291.6	Activation	465.	1.84E-02	0.61
cc	Co-60	1332.5	Activation	7265.	3.03E-01	9.99
dd	Cs-134	1365.1	Fission	423.	1.83E-02	0.60
ff	K-40	1460.8	Other	18.	8.70E-04	0.03
ii	Ba-140	1596.5	Fission	115.	6.43E-03	0.21
jj	Co-58	1674.7	Activation	152.	9.21E-03	0.30

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
1.45E+00	1.18E+00	3.97E-01	3.03E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0    20    40    60    80   100
                                !    !     !     !     !    !
ACTIVATION PRODUCTS:    48.04   ***********************
FISSION PRODUCTS:       38.84   ******************
OTHER PRODUCTION MEANS: 13.12   ******
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-58	Activation	7.30E-01	24.09	50.15
Co-60	Activation	5.80E-01	19.15	39.87
Mn-54	Activation	7.89E-02	2.61	5.43
Fe-59	Activation	4.30E-02	1.42	2.95
Cr-51	Activation	1.25E-02	0.41	0.86
Sn-113	Activation	9.91E-03	0.33	0.68
Ag-110m	Activation	8.00E-04	0.03	0.06
Cs-134	Fission	5.64E-01	18.62	47.94
Cs-137	Fission	3.12E-01	10.31	26.53
Nb-95	Fission	1.83E-01	6.06	15.59
Zr-95	Fission	1.09E-01	3.59	9.24
Ba-140	Fission	6.43E-03	0.21	0.55
Ru-103	Fission	1.80E-03	0.06	0.15
UNKNOWN	Other	2.55E-01	8.43	64.28
ANNIH	Other	1.41E-01	4.66	35.50
K-40	Other	8.70E-04	0.03	0.22

****

****

PLANT: UNIT B PENETRATION BUILDING

LOCATION: FUEL POOL HEAT EXCHANGER

FILE NAME: POM18 GEOMETRIC CORRECTION FACTOR: 1.43E+03

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 59.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 2.60E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	26361.	2.99E-01	11.36
j	ANNIH	511.0	Other	15860.	7.55E-02	2.87
k	Cs-134	563.3	Fission	4886.	2.75E-02	1.04
l	Cs-134	569.3	Fission	9296.	5.33E-02	2.02
m	Cs-134	604.7	Fission	57477.	3.65E-01	13.85
n	Cs-137	661.6	Fission	65610.	4.84E-01	18.37
o	Zr-95	724.2	Fission	1020.	8.73E-03	0.33
p	Zr-95	756.7	Fission	1426.	1.31E-02	0.50
q	Nb-95	765.8	Fission	5150.	4.83E-02	1.83
r	Cs-134	795.8	Fission	41176.	4.11E-01	15.62
s	Co-58	810.8	Activation	48682.	5.01E-01	19.03
t	Mn-54	834.8	Activation	2011.	2.17E-02	0.82
u	Co-58	863.9	Activation	4.	4.56E-05	0.00
v	Ag-110m	884.7	Activation	361.	4.28E-03	0.16
x	Cs-134	1038.5	Fission	605.	9.27E-03	0.35
y	Fe-59	1099.3	Activation	441.	7.40E-03	0.28
aa	Co-60	1173.2	Activation	6015.	1.12E-01	4.25
bb	Fe-59	1291.6	Activation	220.	4.78E-03	0.18
cc	Co-60	1332.5	Activation	5192.	1.19E-01	4.50
dd	Cs-134	1365.1	Fission	1553.	3.69E-02	1.40
ee	Ag-110m	1384.3	Activation	38.	9.23E-04	0.04
ff	K-40	1460.8	Other	176.	4.66E-03	0.18
gg	Ag-110m	1505.0	Activation	139.	3.87E-03	0.15
ii	Ba-140	1596.5	Fission	480.	1.47E-02	0.56
jj	Co-58	1674.7	Activation	238.	7.90E-03	0.30

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
7.83E-01	1.47E+00	3.79E-01	2.63E+00

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !        !        !        !        !        !
ACTIVATION PRODUCTS:    29.72   ************
FISSION PRODUCTS:       55.88   ************************
OTHER PRODUCTION MEANS: 14.40   ******
```

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-58	Activation	5.09E-01	19.34	65.06
Co-60	Activation	2.30E-01	8.75	29.45
Mn-54	Activation	2.17E-02	0.82	2.77
Fe-59	Activation	1.22E-02	0.46	1.56
Ag-110m	Activation	9.07E-03	0.34	1.16
Cs-134	Fission	9.03E-01	34.28	61.35
Cs-137	Fission	4.84E-01	18.37	32.88
Nb-95	Fission	4.83E-02	1.83	3.28
Zr-95	Fission	2.19E-02	0.83	1.49
Ba-140	Fission	1.47E-02	0.56	1.00
UNKNOWN	Other	2.99E-01	11.36	78.85
ANNIH	Other	7.55E-02	2.87	19.92
K-40	Other	4.66E-03	0.18	1.23

****

PLANT: UNIT B PENETRATION BUILDING

LOCATION: FUEL POOL HEAT EXCHANGER - MAIN LINE

FILE NAME: POM19 GEOMETRIC CORRECTION FACTOR: 1.72E+03

LIVE COUNTING TIME (SEC): 100. DEAD TIME (%): 57.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 7.00E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	UNKNOWN	29.4	Other	9617.	2.71E-01	3.86
c	Cr-51	320.1	Activation	6848.	3.43E-02	0.49
e	Sn-113	391.7	Activation	1803.	1.33E-02	0.19
i	Ru-103	497.1	Fission	1384.	1.52E-02	0.22
j	ANNIH	511.0	Other	26793.	3.17E-01	4.51
k	Cs-134	563.3	Fission	2744.	3.84E-02	0.55
l	Cs-134	569.3	Fission	4230.	6.02E-02	0.86
m	Cs-134	604.7	Fission	27706.	4.37E-01	6.21
n	Cs-137	661.6	Fission	30463.	5.58E-01	7.94
o	Zr-95	724.2	Fission	10011.	2.13E-01	3.03
p	Zr-95	756.7	Fission	8304.	1.90E-01	2.70
q	Nb-95	765.8	Fission	29652.	6.91E-01	9.84
r	Cs-134	795.8	Fission	16475.	4.09E-01	5.82
s	Co-58	810.8	Activation	65292.	1.67E+00	23.77
t	Mn-54	834.8	Activation	8270.	2.22E-01	3.16
y	Fe-59	1099.3	Activation	2095.	8.73E-02	1.24
z	Zn-65	1115.5	Activation	21.	8.96E-04	0.01
aa	Co-60	1173.2	Activation	16942.	7.84E-01	11.15
bb	Fe-59	1291.6	Activation	1368.	7.38E-02	1.05
cc	Co-60	1332.5	Activation	15459.	8.77E-01	12.47
dd	Cs-134	1365.1	Fission	571.	3.37E-02	0.48
ii	Ba-140	1596.5	Fission	212.	1.62E-02	0.23
jj	Co-58	1674.7	Activation	203.	1.68E-02	0.24

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.78E+00	2.66E+00	5.88E-01	7.03E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !       !        !        !        !        !
ACTIVATION PRODUCTS:    53.76   ■■■
```

APPENDIX D

ESTIMATED ATTRIBUTABLE EXPOSURE RATES

AFTER REACTOR SHUTDOWN

This appendix presents the calculations of the estimated attributable exposure rates resulting from activation and fission products after reactor shutdown. The initial radionuclide concentrations are those measured during routine operations inside the primary pipe penetration room as documented in NUREG/CR-4485 Vol. 1. The calculations are presented for times after reactor shutdown of 2, 5, 10, 20, 40, and 80 days. An additional calculation is presented which represents the outage measurement taken 82 days after shutdown with the apparent low-energy x rays omitted.

****

PLANT: UNIT B - DURING ROUTINE OPERATION

LOCATION: INSIDE PRIMARY PIPE PENETRATION ROOM

FILE NAME: GSM13 GEOMETRIC CORRECTION FACTOR: 9.10E+02

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 24.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 3.50E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	Xe-133	81.0	Fission	10056.	8.11E-03	0.23
b	Kr-88	196.3	Fission	600.	7.14E-04	0.02
c	Xe-135	249.8	Fission	17503.	3.30E-02	0.94
d	Cr-51	320.1	Activation	104.	3.29E-04	0.01
e	Cd-115	336.3	Activation	252.	8.77E-04	0.03
f	I-131	364.5	Fission	2204.	8.94E-03	0.26
g	Cr-51	391.8	Activation	339.	1.57E-03	0.04
h	Cs-134	475.4	Fission	2315.	1.50E-02	0.43
i	ANNIH	511.0	Other	3748.	2.80E-02	0.80
j	I-133	529.9	Fission	1152.	9.16E-03	0.26
k	Cs-134	563.3	Fission	5570.	4.91E-02	1.40
l	Cs-134	569.3	Fission	10279.	9.23E-02	2.64
m	Cs-134	604.7	Fission	61689.	6.14E-01	17.53
n	Cs-137	661.6	Fission	97439.	1.13E+00	32.19
o	I-132	667.7	Fission	1111.	1.30E-02	0.37
p	Zr-95	724.2	Fission	89.	1.19E-03	0.03
q	Zr-95	756.7	Fission	260.	3.75E-03	0.11
r	Nb-95	765.8	Fission	1267.	1.86E-02	0.53
s	I-132	772.6	Fission	140.	2.09E-03	0.06
t	Cs-134	795.8	Fission	43320.	6.78E-01	19.38
u	Cs-134	801.8	Fission	4743.	7.52E-02	2.15
v	Co-58	810.8	Activation	6153.	9.93E-02	2.84
w	Kr-88	834.8	Fission	2444.	4.14E-02	1.18
x	Kr-88	898.0	Fission	618.	1.18E-02	0.34
y	Cs-138	1009.8	Fission	224.	5.14E-03	0.15
z	Cs-134	1038.5	Fission	412.	9.90E-03	0.28
aa	I-135	1038.8	Fission	50.	1.20E-03	0.03
bb	Fe-59	1099.3	Activation	313.	8.23E-03	0.24
cc	I-135	1131.5	Fission	232.	6.39E-03	0.18

dd	Cs-134	1167.9	Fission	595.	1.72E-02	0.49
ee	Co-60	1173.2	Activation	4703.	1.37E-01	3.92
ff	I-133	1236.4	Fission	45.	1.43E-03	0.04
gg	I-135	1260.4	Fission	174.	5.69E-03	0.16
hh	Fe-59	1291.6	Activation	246.	8.37E-03	0.24
ii	Co-60	1332.5	Activation	4396.	1.57E-01	4.49
jj	Cs-134	1365.1	Fission	914.	3.40E-02	0.97
kk	Na-24	1368.5	Activation	360.	1.34E-02	0.38
ll	Cs-138	1435.9	Fission	336.	1.36E-02	0.39
mm	I-135	1457.6	Fission	57.	2.36E-03	0.07
nn	K-40	1461.5	Other	128.	5.32E-03	0.15
oo	Ni-65	1481.8	Activation	37.	1.57E-03	0.04
pp	Kr-88	1529.8	Fission	127.	5.69E-03	0.16
qq	Ba-140	1596.5	Fission	86.	4.13E-03	0.12
rr	I-135	1678.0	Fission	61.	3.19E-03	0.09
ss	Sb-124	1691.0	Activation	33.	1.75E-03	0.05
tt	Xe-138	1767.6	Fission	34.	1.94E-03	0.06
uu	Kr-88	1836.0	Fission	598.	3.64E-02	1.04
vv	Kr-88	2029.8	Fission	124.	9.00E-03	0.26
ww	Kr-88	2035.4	Fission	67.	4.89E-03	0.14
xx	Kr-88	2195.8	Fission	123.	1.03E-02	0.29
yy	Kr-88	2392.1	Fission	358.	3.52E-02	1.00
zz	Kr-88	2677.9	Fission	62.	7.61E-03	0.22
aaa	Na-24	2753.9	Activation	152.	1.98E-02	0.56

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
4.50E-01	3.02E+00	3.33E-02	3.50E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !      !       !       !       !       !
ACTIVATION PRODUCTS:    12.85   *****
FISSION PRODUCTS:       86.20   **********************************
OTHER PRODUCTION MEANS:  0.95
```

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.95E-01	8.41	65.49
Co-58	Activation	9.93E-02	2.84	22.08
Na-24	Activation	3.32E-02	0.95	7.38
Fe-59	Activation	1.66E-02	0.47	3.69
Cr-51	Activation	1.90E-03	0.05	0.42
Sb-124	Activation	1.75E-03	0.05	0.39
Ni-65	Activation	1.57E-03	0.04	0.35
Cd-115	Activation	8.77E-04	0.03	0.19
Cs-134	Fission	1.58E+00	45.28	52.52
Cs-137	Fission	1.13E+00	32.19	37.34
Kr-88	Fission	1.63E-01	4.65	5.40
Xe-135	Fission	3.30E-02	0.94	1.09
I-135	Fission	1.88E-02	0.54	0.62
Cs-138	Fission	1.87E-02	0.53	0.62
Nb-95	Fission	1.86E-02	0.53	0.62
I-132	Fission	1.51E-02	0.43	0.50
I-133	Fission	1.06E-02	0.30	0.35
I-131	Fission	8.94E-03	0.26	0.30
Xe-133	Fission	8.11E-03	0.23	0.27
Zr-95	Fission	4.95E-03	0.14	0.16
Ba-140	Fission	4.13E-03	0.12	0.14
Xe-138	Fission	1.94E-03	0.06	0.06
ANNIH	Other	2.80E-02	0.80	84.02
K-40	Other	5.32E-03	0.15	15.98

****

PLANT: UNIT B - ESTIMATED FOR TIME = SHUTDOWN + 2 DAYS

LOCATION: INSIDE PRIMARY PIPE PENETRATION ROOM

FILE NAME: OUTAGE02 GEOMETRIC CORRECTION FACTOR: 9.11E+02

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 24.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 3.19E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	Xe-133	81.0	Fission	7738.	6.23E-03	0.20
c	Xe-135	249.8	Fission	456.	8.58E-04	0.03
d	Cr-51	320.1	Activation	99.	3.12E-04	0.01
e	Cd-115	336.3	Activation	135.	4.69E-04	0.01
f	I-131	364.5	Fission	1855.	7.51E-03	0.24
g	Cr-51	391.8	Activation	323.	1.50E-03	0.05
h	Cs-134	475.4	Fission	2311.	1.49E-02	0.47
i	ANNIH	511.0	Other	2932.	2.19E-02	0.69
j	I-133	529.9	Fission	226.	1.79E-03	0.06
k	Cs-134	563.3	Fission	5560.	4.90E-02	1.54
l	Cs-134	569.3	Fission	10260.	9.20E-02	2.88
m	Cs-134	604.7	Fission	61575.	6.11E-01	19.17
n	Cs-137	661.6	Fission	97427.	1.12E+00	35.25
p	Zr-95	724.2	Fission	87.	1.17E-03	0.04
q	Zr-95	756.7	Fission	255.	3.67E-03	0.12
r	Nb-95	765.8	Fission	1218.	1.79E-02	0.56
t	Cs-134	795.8	Fission	43240.	6.76E-01	21.19
u	Cs-134	801.8	Fission	4734.	7.49E-02	2.35
v	Co-58	810.8	Activation	6034.	9.72E-02	3.05
z	Cs-134	1038.5	Fission	411.	9.86E-03	0.31
bb	Fe-59	1099.3	Activation	304.	7.98E-03	0.25
cc	I-135	1131.5	Fission	2.	5.50E-05	0.00
dd	Cs-134	1167.9	Fission	594.	1.72E-02	0.54
ee	Co-60	1173.2	Activation	4700.	1.37E-01	4.29
ff	I-133	1236.4	Fission	9.	2.85E-04	0.01
gg	I-135	1260.4	Fission	34.	1.11E-03	0.03
hh	Fe-59	1291.6	Activation	239.	8.12E-03	0.25
ii	Co-60	1332.5	Activation	4393.	1.57E-01	4.92
jj	Cs-134	1365.1	Fission	912.	3.39E-02	1.06
kk	Na-24	1368.5	Activation	40.	1.49E-03	0.05
nn	K-40	1461.5	Other	128.	5.31E-03	0.17
qq	Ba-140	1596.5	Fission	77.	3.70E-03	0.12
ss	Sb-124	1691.0	Activation	32.	1.69E-03	0.05
aaa	Na-24	2753.9	Activation	17.	2.21E-03	0.07

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
4.15E-01	2.75E+00	2.72E-02	3.19E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !       !        !        !        !        !
ACTIVATION PRODUCTS:    13.00   *****
FISSION PRODUCTS:       86.14   **********************************
OTHER PRODUCTION MEANS:  0.85
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.94E-01	9.21	70.83
Co-58	Activation	9.72E-02	3.05	23.44
Fe-59	Activation	1.61E-02	0.50	3.88
Na-24	Activation	3.70E-03	0.12	0.89
Cr-51	Activation	1.81E-03	0.06	0.44
Sb-124	Activation	1.69E-03	0.05	0.41
Cd-115	Activation	4.69E-04	0.01	0.11
Cs-134	Fission	1.58E+00	49.50	57.47
Cs-137	Fission	1.12E+00	35.25	40.92
Nb-95	Fission	1.79E-02	0.56	0.65
I-131	Fission	7.51E-03	0.24	0.27
Xe-133	Fission	6.23E-03	0.20	0.23
Zr-95	Fission	4.84E-03	0.15	0.18
Ba-140	Fission	3.70E-03	0.12	0.13
I-133	Fission	2.08E-03	0.07	0.08
I-135	Fission	1.17E-03	0.04	0.04
Xe-135	Fission	8.58E-04	0.03	0.03
ANNIH	Other	2.19E-02	0.69	80.45
K-40	Other	5.31E-03	0.17	19.55

****

****

PLANT: UNIT B - ESTIMATED FOR TIME = SHUTDOWN + 5 DAYS

LOCATION: INSIDE PRIMARY PIPE PENETRATION ROOM

FILE NAME: OUTAGE05 GEOMETRIC CORRECTION FACTOR: 9.11E+02

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 24.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 3.17E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	Xe-133	81.0	Fission	5223.	4.21E-03	0.13
c	Xe-135	249.8	Fission	2.	3.76E-06	0.00
d	Cr-51	320.1	Activation	92.	2.91E-04	0.01
e	Cd-115	336.3	Activation	53.	1.84E-04	0.01
f	I-131	364.5	Fission	1433.	5.81E-03	0.18
g	Cr-51	391.8	Activation	299.	1.39E-03	0.04
h	Cs-134	475.4	Fission	2304.	1.49E-02	0.47
i	ANNIH	511.0	Other	2895.	2.16E-02	0.68
j	I-133	529.9	Fission	20.	1.59E-04	0.01
k	Cs-134	563.3	Fission	5544.	4.89E-02	1.54
l	Cs-134	569.3	Fission	10232.	9.18E-02	2.90
m	Cs-134	604.7	Fission	61404.	6.10E-01	19.25
n	Cs-137	661.6	Fission	97408.	1.13E+00	35.49
p	Zr-95	724.2	Fission	84.	1.13E-03	0.04
q	Zr-95	756.7	Fission	247.	3.56E-03	0.11
r	Nb-95	765.8	Fission	1148.	1.69E-02	0.53
t	Cs-134	795.8	Fission	43120.	6.75E-01	21.28
u	Cs-134	801.8	Fission	4721.	7.48E-02	2.36
v	Co-58	810.8	Activation	5860.	9.45E-02	2.98
z	Cs-134	1038.5	Fission	410.	9.84E-03	0.31
bb	Fe-59	1099.3	Activation	290.	7.62E-03	0.24
dd	Cs-134	1167.9	Fission	592.	1.71E-02	0.54
ee	Co-60	1173.2	Activation	4694.	1.37E-01	4.32
ff	I-133	1236.4	Fission	1.	3.17E-05	0.00
gg	I-135	1260.4	Fission	3.	9.81E-05	0.00
hh	Fe-59	1291.6	Activation	228.	7.75E-03	0.24
ii	Co-60	1332.5	Activation	4388.	1.57E-01	4.95
jj	Cs-134	1365.1	Fission	910.	3.38E-02	1.07
kk	Na-24	1368.5	Activation	1.	3.73E-05	0.00
nn	K-40	1461.5	Other	128.	5.31E-03	0.17
qq	Ba-140	1596.5	Fission	66.	3.17E-03	0.10
ss	Sb-124	1691.0	Activation	31.	1.64E-03	0.05
aaa	Na-24	2753.9	Activation	1.	1.30E-04	0.00

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
4.07E-01	2.74E+00	2.69E-02	3.17E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !       !       !       !       !       !
ACTIVATION PRODUCTS:     12.85  *****
FISSION PRODUCTS:        86.31  ************************************
OTHER PRODUCTION MEANS:   0.85
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.94E-01	9.26	72.12
Co-58	Activation	9.45E-02	2.98	23.21
Fe-59	Activation	1.54E-02	0.48	3.77
Cr-51	Activation	1.68E-03	0.05	0.41
Sb-124	Activation	1.64E-03	0.05	0.40
Cd-115	Activation	1.84E-04	0.01	0.05
Na-24	Activation	1.67E-04	0.01	0.04
Cs-134	Fission	1.58E+00	49.71	57.60
Cs-137	Fission	1.13E+00	35.49	41.12
Nb-95	Fission	1.69E-02	0.53	0.62
I-131	Fission	5.81E-03	0.18	0.21
Zr-95	Fission	4.69E-03	0.15	0.17
Xe-133	Fission	4.21E-03	0.13	0.15
Ba-140	Fission	3.17E-03	0.10	0.12
I-133	Fission	1.90E-04	0.01	0.01
I-135	Fission	9.81E-05	0.00	0.00
Xe-135	Fission	3.76E-06	0.00	0.00
ANNIH	Other	2.16E-02	0.68	80.25
K-40	Other	5.31E-03	0.17	19.75

****

****

PLANT: UNIT B - ESTIMATED FOR TIME = SHUTDOWN + 10 DAYS

LOCATION: INSIDE PRIMARY PIPE PENETRATION ROOM

FILE NAME: OUTAGE10 GEOMETRIC CORRECTION FACTOR: 9.10E+02

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 24.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 3.15E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	Xe-133	81.0	Fission	2713.	2.19E-03	0.07
d	Cr-51	320.1	Activation	81.	2.56E-04	0.01
e	Cd-115	336.3	Activation	11.	3.83E-05	0.00
f	I-131	364.5	Fission	932.	3.78E-03	0.12
g	Cr-51	391.8	Activation	264.	1.22E-03	0.04
h	Cs-134	475.4	Fission	2294.	1.48E-02	0.47
i	ANNIH	511.0	Other	2874.	2.14E-02	0.68
k	Cs-134	563.3	Fission	5519.	4.87E-02	1.54
l	Cs-134	569.3	Fission	10184.	9.14E-02	2.90
m	Cs-134	604.7	Fission	61120.	6.08E-01	19.29
n	Cs-137	661.6	Fission	97377.	1.13E+00	35.72
p	Zr-95	724.2	Fission	80.	1.07E-03	0.03
q	Zr-95	756.7	Fission	234.	3.37E-03	0.11
r	Nb-95	765.8	Fission	1039.	1.53E-02	0.48
t	Cs-134	795.8	Fission	42921.	6.72E-01	21.33
u	Cs-134	801.8	Fission	4699.	7.44E-02	2.36
v	Co-58	810.8	Activation	5581.	9.00E-02	2.86
z	Cs-134	1038.5	Fission	408.	9.79E-03	0.31
bb	Fe-59	1099.3	Activation	268.	7.04E-03	0.22
dd	Cs-134	1167.9	Fission	590.	1.71E-02	0.54
ee	Co-60	1173.2	Activation	4685.	1.37E-01	4.34
hh	Fe-59	1291.6	Activation	211.	7.18E-03	0.23
ii	Co-60	1332.5	Activation	4380.	1.57E-01	4.97
jj	Cs-134	1365.1	Fission	906.	3.37E-02	1.07
nn	K-40	1461.5	Other	128.	5.32E-03	0.17
qq	Ba-140	1596.5	Fission	50.	2.40E-03	0.08
ss	Sb-124	1691.0	Activation	29.	1.53E-03	0.05

********** **********

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
4.01E-01	2.72E+00	2.68E-02	3.15E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !      !       !       !       !       !
ACTIVATION PRODUCTS:    12.72   *****
FISSION PRODUCTS:       86.43   ************************************
OTHER PRODUCTION MEANS:  0.85
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.93E-01	9.31	73.21
Co-58	Activation	9.00E-02	2.86	22.47
Fe-59	Activation	1.42E-02	0.45	3.55
Sb-124	Activation	1.53E-03	0.05	0.38
Cr-51	Activation	1.48E-03	0.05	0.37
Cd-115	Activation	3.83E-05	0.00	0.01
Cs-134	Fission	1.57E+00	49.82	57.64
Cs-137	Fission	1.13E+00	35.72	41.33
Nb-95	Fission	1.53E-02	0.48	0.56
Zr-95	Fission	4.45E-03	0.14	0.16
I-131	Fission	3.78E-03	0.12	0.14
Ba-140	Fission	2.40E-03	0.08	0.09
Xe-133	Fission	2.19E-03	0.07	0.08
ANNIH	Other	2.14E-02	0.68	80.13
K-40	Other	5.32E-03	0.17	19.87

****

****

PLANT: UNIT B - ESTIMATED FOR TIME = SHUTDOWN + 20 DAYS

LOCATION: INSIDE PRIMARY PIPE PENETRATION ROOM

FILE NAME: OUTAGE20 GEOMETRIC CORRECTION FACTOR: 9.10E+02

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 24.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 3.11E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	Xe-133	81.0	Fission	732.	5.91E-04	0.02
d	Cr-51	320.1	Activation	63.	1.99E-04	0.01
e	Cd-115	336.3	Activation	1.	3.48E-06	0.00
f	I-131	364.5	Fission	394.	1.60E-03	0.05
g	Cr-51	391.8	Activation	206.	9.56E-04	0.03
h	Cs-134	475.4	Fission	2273.	1.47E-02	0.47
i	ANNIH	511.0	Other	2838.	2.12E-02	0.68
k	Cs-134	563.3	Fission	5468.	4.82E-02	1.55
l	Cs-134	569.3	Fission	10090.	9.06E-02	2.91
m	Cs-134	604.7	Fission	60557.	6.02E-01	19.37
n	Cs-137	661.6	Fission	97316.	1.13E+00	36.18
p	Zr-95	724.2	Fission	72.	9.67E-04	0.03
q	Zr-95	756.7	Fission	210.	3.03E-03	0.10
r	Nb-95	765.8	Fission	853.	1.25E-02	0.40
t	Cs-134	795.8	Fission	42525.	6.66E-01	21.42
u	Cs-134	801.8	Fission	4656.	7.38E-02	2.37
v	Co-58	810.8	Activation	4642.	7.49E-02	2.41
z	Cs-134	1038.5	Fission	404.	9.71E-03	0.31
bb	Fe-59	1099.3	Activation	230.	6.05E-03	0.19
dd	Cs-134	1167.9	Fission	584.	1.69E-02	0.54
ee	Co-60	1173.2	Activation	4669.	1.36E-01	4.38
hh	Fe-59	1291.6	Activation	181.	6.16E-03	0.20
ii	Co-60	1332.5	Activation	4364.	1.56E-01	5.02
jj	Cs-134	1365.1	Fission	897.	3.34E-02	1.07
nn	K-40	1461.5	Other	128.	5.32E-03	0.17
qq	Ba-140	1596.5	Fission	29.	1.39E-03	0.04
ss	Sb-124	1691.0	Activation	26.	1.38E-03	0.04

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.82E-01	2.70E+00	2.65E-02	3.11E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !      !       !       !       !       !
ACTIVATION PRODUCTS:    12.29   *****
FISSION PRODUCTS:       86.86   ***********************************
OTHER PRODUCTION MEANS:  0.85
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.92E-01	9.40	76.53
Co-58	Activation	7.49E-02	2.41	19.61
Fe-59	Activation	1.22E-02	0.39	3.20
Sb-124	Activation	1.38E-03	0.04	0.36
Cr-51	Activation	1.15E-03	0.04	0.30
Cd-115	Activation	3.48E-06	0.00	0.00
Cs-134	Fission	1.56E+00	50.03	57.60
Cs-137*	Fission	1.13E+00	36.18	41.66
Nb-95	Fission	1.25E-02	0.40	0.46
Zr-95	Fission	4.00E-03	0.13	0.15
I-131	Fission	1.60E-03	0.05	0.06
Ba-140	Fission	1.39E-03	0.04	0.05
Xe-133	Fission	5.91E-04	0.02	0.02
ANNIH	Other	2.12E-02	0.68	79.93
K-40	Other	5.32E-03	0.17	20.07

****

****

PLANT: UNIT B - ESTIMATED FOR TIME = SHUTDOWN + 40 DAYS

LOCATION: INSIDE PRIMARY PIPE PENETRATION ROOM

FILE NAME: OUTAGE40 GEOMETRIC CORRECTION FACTOR: 9.09E+02

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 24.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 3.06E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
a	Xe-133	81.0	Fission	53.	4.28E-05	0.00
d	Cr-51	320.1	Activation	38.	1.20E-04	0.00
f	I-131	364.5	Fission	70.	2.84E-04	0.01
g	Cr-51	391.8	Activation	125.	5.80E-04	0.02
h	Cs-134	475.4	Fission	2231.	1.45E-02	0.47
i	ANNIH	511.0	Other	2780.	2.08E-02	0.68
k	Cs-134	563.3	Fission	5367.	4.74E-02	1.55
l	Cs-134	569.3	Fission	9905.	8.90E-02	2.91
m	Cs-134	604.7	Fission	59445.	5.92E-01	19.34
n	Cs-137	661.6	Fission	97193.	1.12E+00	36.75
p	Zr-95	724.2	Fission	58.	7.79E-04	0.03
q	Zr-95	756.7	Fission	170.	2.45E-03	0.08
r	Nb-95	765.8	Fission	574.	8.45E-03	0.28
t	Cs-134	795.8	Fission	41744.	6.54E-01	21.38
u	Cs-134	801.8	Fission	4570.	7.25E-02	2.37
v	Co-58	810.8	Activation	4164.	6.73E-02	2.20
z	Cs-134	1038.5	Fission	397.	9.54E-03	0.31
bb	Fe-59	1099.3	Activation	169.	4.45E-03	0.15
dd	Cs-134	1167.9	Fission	573.	1.66E-02	0.54
ee	Co-60	1173.2	Activation	4635.	1.35E-01	4.42
hh	Fe-59	1291.6	Activation	133.	4.53E-03	0.15
ii	Co-60	1332.5	Activation	4333.	1.55E-01	5.07
jj	Cs-134	1365.1	Fission	881.	3.28E-02	1.07
nn	K-40	1461.5	Other	128.	5.32E-03	0.17
qq	Ba-140	1596.5	Fission	10.	4.81E-04	0.02
ss	Sb-124	1691.0	Activation	21.	1.11E-03	0.04

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.69E-01	2.67E+00	2.61E-02	3.06E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0      20      40      60      80      100
                                !      !       !       !       !       !
ACTIVATION PRODUCTS:    12.04   *****
FISSION PRODUCTS:       87.10   ************************************
OTHER PRODUCTION MEANS:  0.85
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.91E-01	9.49	78.82
Co-58	Activation	6.73E-02	2.20	18.25
Fe-59	Activation	8.98E-03	0.29	2.44
Sb-124	Activation	1.11E-03	0.04	0.30
Cr-51	Activation	7.00E-04	0.02	0.19
Cs-134	Fission	1.53E+00	49.94	57.34
Cs-137	Fission	1.12E+00	36.75	42.19
Nb-95	Fission	8.45E-03	0.28	0.32
Zr-95	Fission	3.23E-03	0.11	0.12
Ba-140	Fission	4.81E-04	0.02	0.02
I-131	Fission	2.84E-04	0.01	0.01
Xe-133	Fission	4.28E-05	0.00	0.00
ANNIH	Other	2.08E-02	0.68	79.60
K-40	Other	5.32E-03	0.17	20.40

****

****

PLANT: UNIT B - ESTIMATED FOR TIME = SHUTDOWN + 80 DAYS

LOCATION: INSIDE PRIMARY PIPE PENETRATION ROOM

FILE NAME: OUTAGE80 GEOMETRIC CORRECTION FACTOR: 9.10E+02

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 24.0

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 2.96E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
d	Cr-51	320.1	Activation	14.	4.42E-05	0.00
f	I-131	364.5	Fission	2.	8.11E-06	0.00
g	Cr-51	391.8	Activation	46.	2.13E-04	0.01
h	Cs-134	475.4	Fission	2150.	1.39E-02	0.47
i	ANNIH	511.0	Other	2692.	2.01E-02	0.68
k	Cs-134	563.3	Fission	5172.	4.56E-02	1.54
l	Cs-134	569.3	Fission	9545.	8.57E-02	2.90
m	Cs-134	604.7	Fission	57283.	5.70E-01	19.24
n	Cs-137	661.6	Fission	96947.	1.12E+00	37.85
p	Zr-95	724.2	Fission	38.	5.10E-04	0.02
q	Zr-95	756.7	Fission	112.	1.61E-03	0.05
r	Nb-95	765.8	Fission	260.	3.82E-03	0.13
t	Cs-134	795.8	Fission	40226.	6.30E-01	21.27
u	Cs-134	801.8	Fission	4404.	6.98E-02	2.36
v	Co-58	810.8	Activation	2818.	4.55E-02	1.54
z	Cs-134	1038.5	Fission	383.	9.20E-03	0.31
bb	Fe-59	1099.3	Activation	91.	2.39E-03	0.08
dd	Cs-134	1167.9	Fission	553.	1.60E-02	0.54
ee	Co-60	1173.2	Activation	4569.	1.33E-01	4.50
hh	Fe-59	1291.6	Activation	72.	2.45E-03	0.08
ii	Co-60	1332.5	Activation	4270.	1.53E-01	5.16
jj	Cs-134	1365.1	Fission	849.	3.16E-02	1.07
nn	K-40	1461.5	Other	128.	5.32E-03	0.18
qq	Ba-140	1596.5	Fission	1.	4.81E-05	0.00
ss	Sb-124	1691.0	Activation	13.	6.88E-04	0.02

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
3.37E-01	2.60E+00	2.54E-02	2.96E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !        !        !        !        !        !
ACTIVATION PRODUCTS:    11.39   *****
FISSION PRODUCTS:       87.75   ***********************************
OTHER PRODUCTION MEANS:  0.86
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	2.86E-01	9.66	84.80
Co-58	Activation	4.55E-02	1.54	13.48
Fe-59	Activation	4.84E-03	0.16	1.44
Sb-124	Activation	6.88E-04	0.02	0.20
Cr-51	Activation	2.58E-04	0.01	0.08
Cs-134	Fission	1.47E+00	49.69	56.63
Cs-137	Fission	1.12E+00	37.85	43.14
Nb-95	Fission	3.82E-03	0.13	0.15
Zr-95	Fission	2.12E-03	0.07	0.08
Ba-140	Fission	4.81E-05	0.00	0.00
I-131	Fission	8.11E-06	0.00	0.00
ANNIH	Other	2.01E-02	0.68	79.07
K-40	Other	5.32E-03	0.18	20.93

****

****

PLANT: UNIT B PENETRATION BUILDING

LOCATION: INSIDE PIPE PENETRATION ROOM

FILE NAME: POM06X GEOMETRIC CORRECTION FACTOR: 4.06E+03

LIVE COUNTING TIME (SEC): 300. DEAD TIME (%): 22.3

IONIZATION CHAMBER MEASMT. OF EXPOSURE RATE AT DETECTOR (mR/hr): 1.70E+00

****

****

*** DATA ANALYSIS ***

PEAK ID	RADIO-NUCLIDE	ENERGY (KeV)	MEANS OF PRODUCTION	TOTAL COUNTS	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE
d	I-131	364.5	Fission	3367.	3.06E-03	0.18
e	Sn-113	391.7	Activation	1705.	1.77E-03	0.10
g	Cs-138	462.8	Fission	581.	8.08E-04	0.05
h	Cs-134	475.4	Fission	3130.	4.55E-03	0.27
j	ANNIH	511.0	Other	6881.	1.15E-02	0.68
k	Cs-134	563.3	Fission	16598.	3.28E-02	1.93
l	Cs-134	569.3	Fission	22736.	4.58E-02	2.69
m	Cs-134	604.7	Fission	157626.	3.52E-01	20.69
n	Cs-137	661.6	Fission	256608.	6.66E-01	39.15
o	Zr-95	724.2	Fission	629.	1.89E-03	0.11
p	Zr-95	756.7	Fission	860.	2.78E-03	0.16
q	Nb-95	765.8	Fission	3232.	1.07E-02	0.63
r	Cs-134	795.8	Fission	89731.	3.15E-01	18.54
s	Co-58	810.8	Activation	14874.	5.39E-02	3.17
t	Mn-54	834.8	Activation	5051.	1.92E-02	1.13
x	Cs-134	1038.5	Fission	864.	4.66E-03	0.27
y	Fe-59	1099.3	Activation	182.	1.07E-03	0.06
aa	Co-60	1173.2	Activation	11329.	7.42E-02	4.36
bb	Fe-59	1291.6	Activation	156.	1.19E-03	0.07
cc	Co-60	1332.5	Activation	9250.	7.43E-02	4.37
dd	Cs-134	1365.1	Fission	2149.	1.79E-02	1.06
ff	K-40	1460.8	Other	78.	7.27E-04	0.04
ii	Ba-140	1596.5	Fission	154.	1.66E-03	0.10
jj	Co-58	1674.7	Activation	56.	6.54E-04	0.04
kk	Sb-124	1691.0	Activation	45.	5.34E-04	0.03
ll	Rn-222	1764.5	Other	22.	2.81E-04	0.02
mm	Co-56	1771.5	Activation	132.	1.70E-03	0.10

****

****

*** ATTRIBUTABLE DOSE RATE (mR/hr) ***

ACTIVATION PRODUCTS	FISSION PRODUCTS	OTHER MEANS OF PRODUCTION	TOTAL EXPOSURE RATE
2.28E-01	1.46E+00	1.25E-02	1.70E+00

****

****

*** PERCENTAGE OF TOTAL EXPOSURE RATE (%) ***

```
                                0       20       40       60       80       100
                                !       !        !        !        !        !
ACTIVATION PRODUCTS:     13.43  *****
FISSION PRODUCTS:        85.83  ************************************
OTHER PRODUCTION MEANS:   0.74
```

****

****

*** RADIONUCLIDE CONTRIBUTIONS ***

RADIO-NUCLIDE	MEANS OF PRODUCTION	EXP. RATE (mR/hr)	% OF TOTAL EXP. RATE	% OF EXP. RATE BY PROD. GROUP
Co-60	Activation	1.48E-01	8.73	64.99
Co-58	Activation	5.45E-02	3.21	23.87
Mn-54	Activation	1.92E-02	1.13	8.40
Fe-59	Activation	2.26E-03	0.13	0.99
Sn-113	Activation	1.77E-03	0.10	0.78
Co-56	Activation	1.70E-03	0.10	0.74
Sb-124	Activation	5.34E-04	0.03	0.23
Cs-134	Fission	7.73E-01	45.45	52.96
Cs-137	Fission	6.66E-01	39.15	45.61
Nb-95	Fission	1.07E-02	0.63	0.73
Zr-95	Fission	4.68E-03	0.28	0.32
I-131	Fission	3.06E-03	0.18	0.21
Ba-140	Fission	1.66E-03	0.10	0.11
Cs-138	Fission	8.08E-04	0.05	0.06
ANNIH	Other	1.15E-02	0.68	91.96
K-40	Other	7.27E-04	0.04	5.80
Rn-222	Other	2.81E-04	0.02	2.24

****

NRC FORM 336 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1 REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG/CR-4485
PNL-5606
Vol. 2

2. TITLE AND SUBTITLE
The Impact of Fuel Cladding Failure Events on Occupational Radiation Exposures at Nuclear Power Plants
Case Study: PWR During an Outage

3 LEAVE BLANK

4 DATE REPORT COMPLETED
MONTH: December
YEAR: 1986

5. AUTHOR(S)
M. P. Moeller, G. F. Martin, J. L. Kenoyer

6 DATE REPORT ISSUED
MONTH: August
YEAR: 1987

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Pacific Northwest Laboratory
P.O. Box 999
Richland, WA 99352

8 PROJECT/TASK/WORK UNIT NUMBER

9 FIN OR GRANT NUMBER
FIN B-2987

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Reactor Accident Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a TYPE OF REPORT
Technical

b PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This report is the second in a series of case studies designed to evaluate the magnitude of increase in occupational radiation exposures at commercial U.S. nuclear power plants resulting from small incidents or abnormal events. The event evaluated is fuel cladding failure, which can result in elevated primary coolant activity and increased radiation exposure rates within a plant. For this case study, radiation measurements were made at a pressurized-water reactor (PWR) during a maintenance and refueling outage. The PWR had been operating for 22 months with fuel cladding failure characterized as 105 pin-hole leakers, the equivalent of 0.212 percent failed fuel. Gamma spectroscopy measurements, radiation exposure rate determinations, thermoluminescent dosimeter (TLD) assessments, and air sample analyses were made in the plant's radwaste, pipe penetration, and containment buildings.

Based on the data collected, evaluations indicate that the relative contributions of activation products and fission products to the total exposure rates were constant over the duration of the outage. This constancy is due to the significant contribution from the longer-lived isotopes of cesium (a fission product) and cobalt (an activation product). For this reason, fuel cladding failure events remain as significant to occupational radiation exposure during an outage as during routine operations. As documented in the previous case study (NUREG/CR-4485 Vol. 1), fuel cladding failure events increased radiation exposure rates an estimated 540 percent in some areas of the plant during routine operations. Consequently, such events can result in significantly greater radiation exposure rates in many areas of the plant during the maintenance and refueling outages than would have been present under normal fuel conditions.

14 DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
Fuel Element Failure - Accidents
Radiation Hazards
Reactor Safety

b. IDENTIFIERS/OPEN-ENDED TERMS
Barrier Degradation Events
Occupational Radiation Exposure

15 AVAILABILITY STATEMENT
Unlimited

16 SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17 NUMBER OF PAGES

18 PRICE

*U.S. GOVERNMENT PRINTING OFFICE:1987-181-682:60206

BEST COPY AVAILABLE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC
PERMIT No. G-67

1

BEST COPY AVAILABLE